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Nuclear Materials & Propulsion Operation

INTRODUCTION TO NUCLEAR PROPULSION

Lectures 17 & 18 - MATERIALS FOR NUCLEAR APPLICATION

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1.0 INTRODUCTION

It is generally accepted, and rightly so, that the advent of nuclear reactors has caused the writing of a completely new chapter in the textbooks of material technology. Not only have new theories of materials been evolved, new handling and fabrication procedures innovated but more importantly, and we might say to our own dismay, new conditions of restraint have been placed on material application. It is almost as though we have been deprived of one degree of freedom in our former and more conventional design and application philosophies. As a matter of fact, some dedicated materials technologists might express the opinion that in this new nuclear phase of our technology we have been deprived of almost all degrees of freedom. This latter comment is more facetious than factful for we have all personally witnessed the successful application of nuclear energy to power reactors as well as propulsion reactors. However, this is not to mitigate in any way the pertinency of the above comments but is more to reflect credit on the ingenuity of man.

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In limiting this lecture to a discussion of nuclear propulsion systems for space application, it might appear that this restriction in itself might lessen the technical complexity of the assignment. Let me assure you, however, that this is not the case. What it does do is eliminate many materials, and I might say many currently operable nuclear systems, from our consideration and causes us to focus our attention on a narrower range of materials and concepts. Whether or not this lessens the complexity of our objective is a moot point.

As another word of introduction I think it is well to emphasize that this discussion today of materials for nuclear propulsion systems in space should not convey the impression that these are the only problems facing the materials engineer. Many problems of equal technical importance both precede and follow the considerations which form the subject matter of this lecture. Such problems as nuclear fuel production and fabrication, alloy development, fuel recovery and reprocessing and material property evaluation, to mention only a few, are all facets of nuclear materials technology and are just as necessary to the success of our nuclear space program as our chosen subject matter. However, limitation of time precludes the assignment of equal consideration to all of these categories and it will suffice for the moment to merely acknowledge their existence.

The condition of restraint referred to above is obviously a nuclear one and requires that considerations of radiation damage and neutron conservation be included in our evaluations of materials for possible reactor application. Both factors must be given early consideration in a program of material selection, for reactor reliability as well as reactor performance efficiency are determined to a great extent by these concepts.

Ideally it is possible to identify a sequence of considerations which should be adhered to in the selection of materials for nuclear propulsion reactors (actually this same list could apply to all nuclear reactors). Such a list would be as follows:

1. Nuclear Considerations - to determine if materials have proper nuclear properties.

2. Temperature Considerations - to determine if materials have proper operating temperature capabilities.

3. Chemical Considerations - to determine if materials have required chemical stability, compatibility, etc.

4. Mechanical Considerations - to determine if materials have required strength.

5. Thermal Considerations - to determine if materials have required thermal conductivity, thermal expansion, etc.

6. Fabrication Considerations - to determine if materials can be fabricated in desired configurations.

The inclusion of the above categories in this discussion is not meant to imply that the selection of materials follows an orderly sequence of events. Such is not the case. As a matter of fact, it cannot even be assumed that the outcome of such an analysis would yield one and only one material which would be perfect in every way for a given application. This list merely serves to delineate the numerous factors which must be considered in such material selection. Experience has shown that many compromises must be made in this process and the final choice, while it might be a perfectly satisfactory material, might not possess the properties we originally hoped to achieve at the start of the material selection process. In summation then it can be said that while we can specify quite explicitly the properties we desire in a material, it might be impossible to identify a single material possessing all these features and characteristics. The final selection then must be made after a proper weighting of all the involved factors. A further complication is represented by the fact that a given material might be ideal for a given application in a certain environment (coolant) and yet be completely unsatisfactory for operation at the same conditions in a different environment.

The application of the above considerations is best illustrated by considering specific instances of material selection for the various components of a reactor. Such components are normally subdivided as follows: fuel elements, moderators and reflectors, controls, shields, and finally structural components. Based on this classification, a discussion of the factors pertinent to

material selection in each of these areas will aid in identifying the major problems confronting any material selection program. In each category of the discussion the purpose of each component will be reviewed, for it will be seen that this specific function exerts a major influence on material selection.

Before proceeding to a discussion of the individual reactor components, it might be well to review the schematic diagram of a typical nuclear reactor shown in Figure 1. Briefly, the neutrons produced by fission of the nuclear fuel in the fuel elements must be moderated (slowed down) to allow them to be captured or absorbed by other fuel atoms to propagate the fission reaction. The heat released in this fission reaction is transferred to the coolant flowing through the reactor core. The reflector functions to conserve neutrons and in effect reflects many neutrons back into the core which would otherwise escape. The shield offers protection to components and personnel outside the reactor core by decreasing the levels of nuclear radiation (neutrons, alpha, beta, and gamma rays) which penetrate the core. The outer structure provides for containment and support.

2.0   FUEL ELEMENTS

The fuel elements are the power producers of the nuclear reactor and it is these components which are responsible for the existence of all the other reactor components. For example, if these reactors were self-controlling, there would be no need for control materials and if they were self-moderating or required no moderation there would be no need for moderators. In view of this important role of the fuel element, it requires very special consideration if very special performance is to be achieved. Furthermore, since the operating temperature of the fuel element is higher than in any other part of the reactor, the selection of these materials is correspondingly more complex.

In general the term, fuel element, applies to an integral structure containing the nuclear fuel itself, any matrix material in which the fuel might be dispersed, and any cladding material which might be employed for corrosion resistance and retention of fission products. Since such a structure employing all three of these aspects will frequently require three different materials, the material selection procedures are necessarily compounded. Further since these individual materials must possess different properties, it will serve a useful purpose if we discuss each segment of this structure separately.
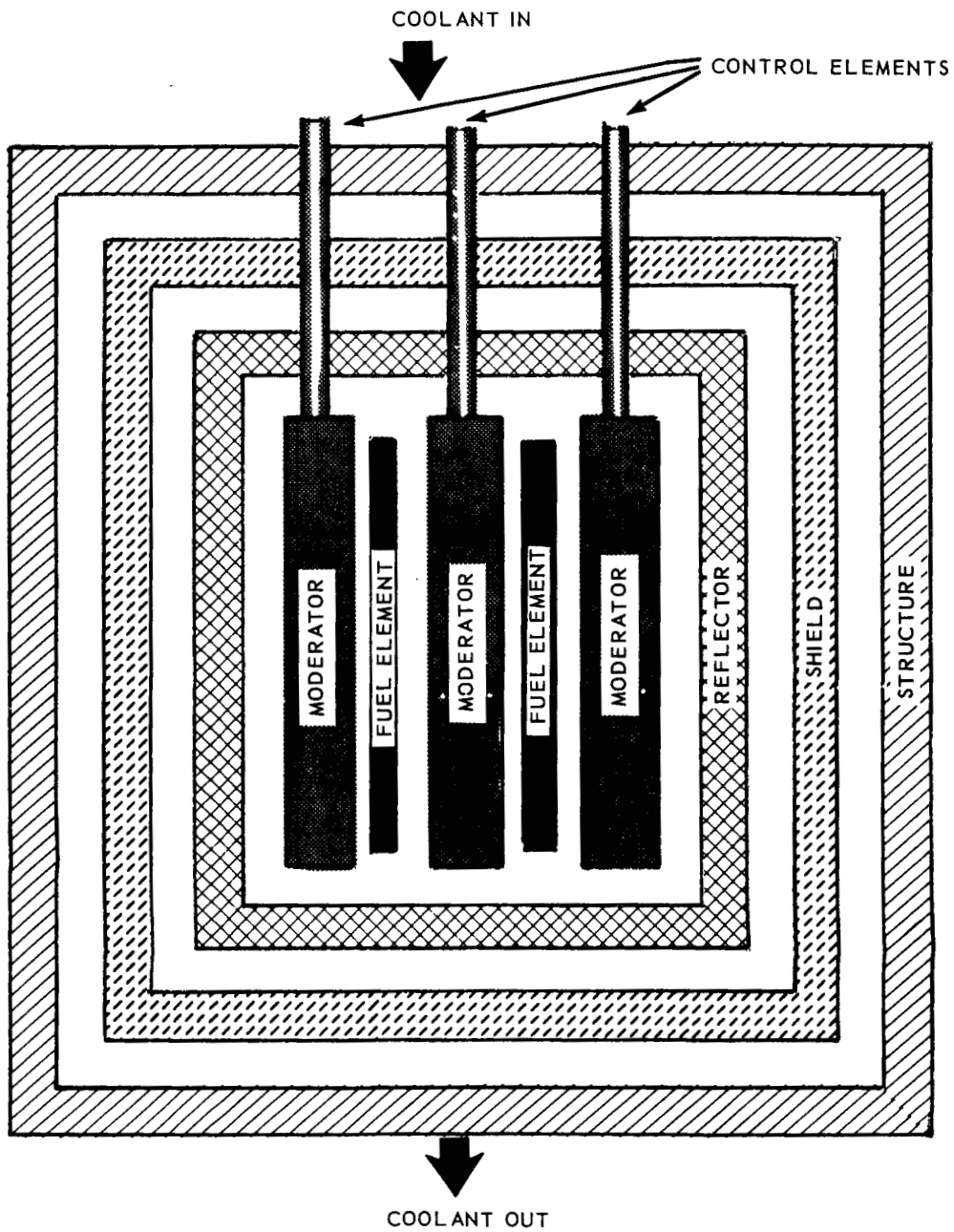
COOLANT IN

CONTROL ELEMENTS

MODERATOR

FUEL ELEMENT

MODERATOR

FUEL ELEMENT

MODERATOR

REFLECTOR

SHIELD

STRUCTURE

COOLANT OUT

Fig. 1 — Schematic of typical nuclear reactor

2.1 Fuel Material

The basic nuclear fuel materials are U^{233}, U^{235} and Pu^{239}
which might at first lead one to believe that the selection of
the fuel material would be relatively straightforward. However,
this is unfortunately not the case, for these species need not be
used in their atomic form, but may be used just as conveniently
and usually more conveniently in alloy or compound forms. The
ability to employ fuels as uranium compounds with other elements
injects an extensive latitude into our material selection pro-
cedures, for the number of possible combinations is quite large.
A list of some of the current uranium compounds (in this dis-
cussion we will confine our remarks to solid fuel materials) now
under consideration is shown in Table 1. The advantage afforded
by these materials in melting point alone is evident when we
consider that the melting point of uranium metal is about $1130°C$.
Since the melting points shown in Table 1 are all essentially
the same, the selection of the material to be used in a given
application is dictated to a great extent by considerations of
compatibility of the fuel material with the matrix and clad
materials.

2.2 Matrix Material

While it is possible to form a fuel element by capsulating
any of the fuel compounds given in Table 1 in a suitable con-
tainer or cladding, this is seldom if ever an acceptable struc-
ture for power reactors of the type under consideration. In
almost every instance it is desirable, and in many cases impera-
tive, that the fuel compound be dispersed in some matrix material.
For example, in the case of UO_2 the thermal conductivity of this
material is so low that the generation of high power density
would lead to such extremely high temperature gradients within
the material that melting might be encountered at the mid-plane
of the matrix. It becomes necessary in such cases to disperse
this material in a continuous matrix of a higher thermal con-
ductivity material to enhance the heat transfer characteristics.
In addition to thermal conductivity, such considerations as low
strength, brittleness and perhaps poor thermal shock resistance
may also suggest such measures. These dispersion type fuel
elements are quite common, and reference is made to the well-
known stainless steel-UO_2 dispersion type and the Nichrome-UO_2
type originally prepared for use in the Aircraft Nuclear Pro-
pulsion Program.

Because of the intimate admixture of fuel and matrix
material and the small particle size of the fuel, the tempera-
tures attained in these materials will be essentially identical.
For this reason it is important to select matrix materials with
temperature stability limits at least equal to those of the
fuel. A review of the melting points of various materials is
a good starting point for such considerations. The melting
points for various materials are shown in Figures 2, 3, 4 and 5.
Note that the highest melting points are exhibited by graphite,
certain refractory metal carbides, and by tungsten and rhenium.

TABLE 1

PHYSICAL PROPERTIES OF SOME TYPICAL URANIUM COMPOUNDS

Compound	Density, g/cm^3	Uranium Density, g/cm^3	Melting Point, $^{\circ}C$
UO_2	10.97	10.5	2800
UC	13.63	12.97	2400
UN	14.32	13.52	2885
UB_2	12.82	11.7	2440
UB_4	9.3	7.9	> 2550
UC_2	11.68	10.61	2400
US	10.87	9.6	2460

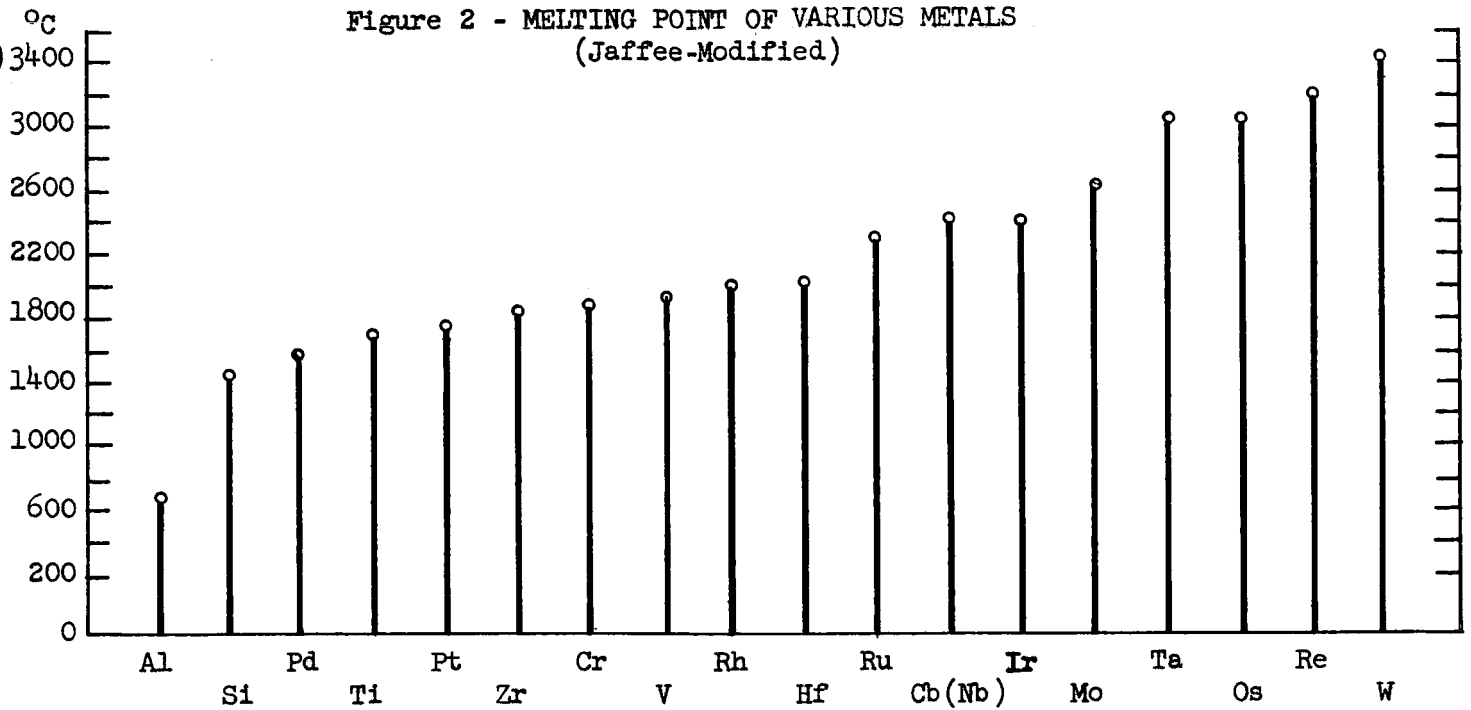Figure 2 - MELTING POINT OF VARIOUS METALS
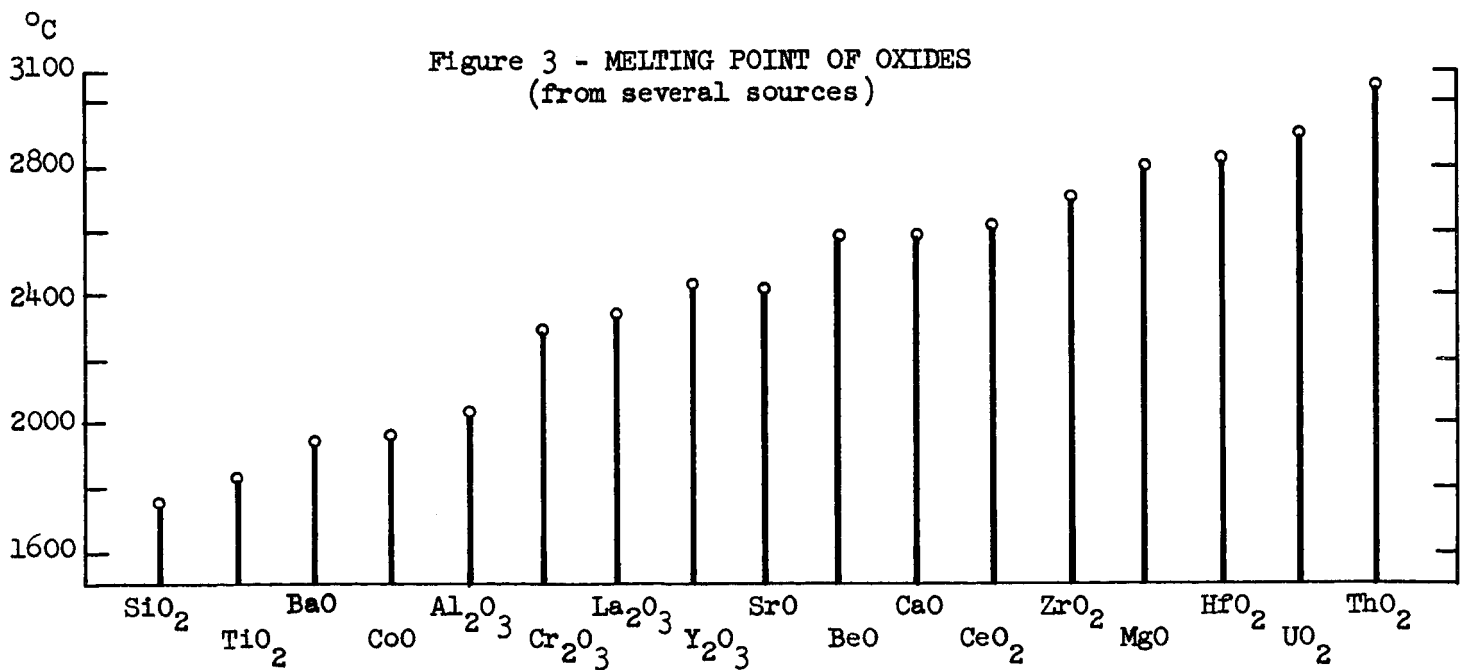(Jaffee-Modified)

Figure 3 - MELTING POINT OF OXIDES
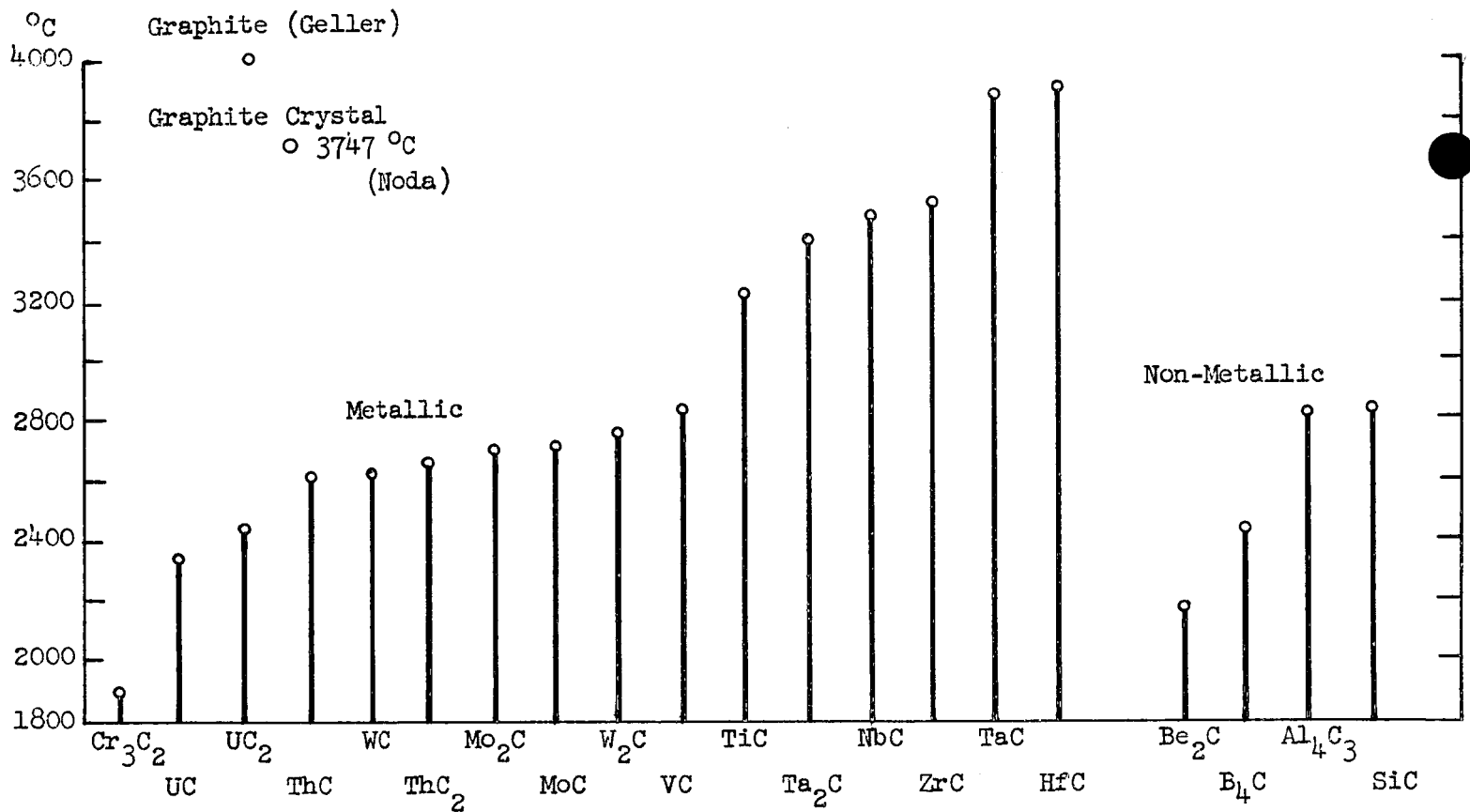(from several sources)

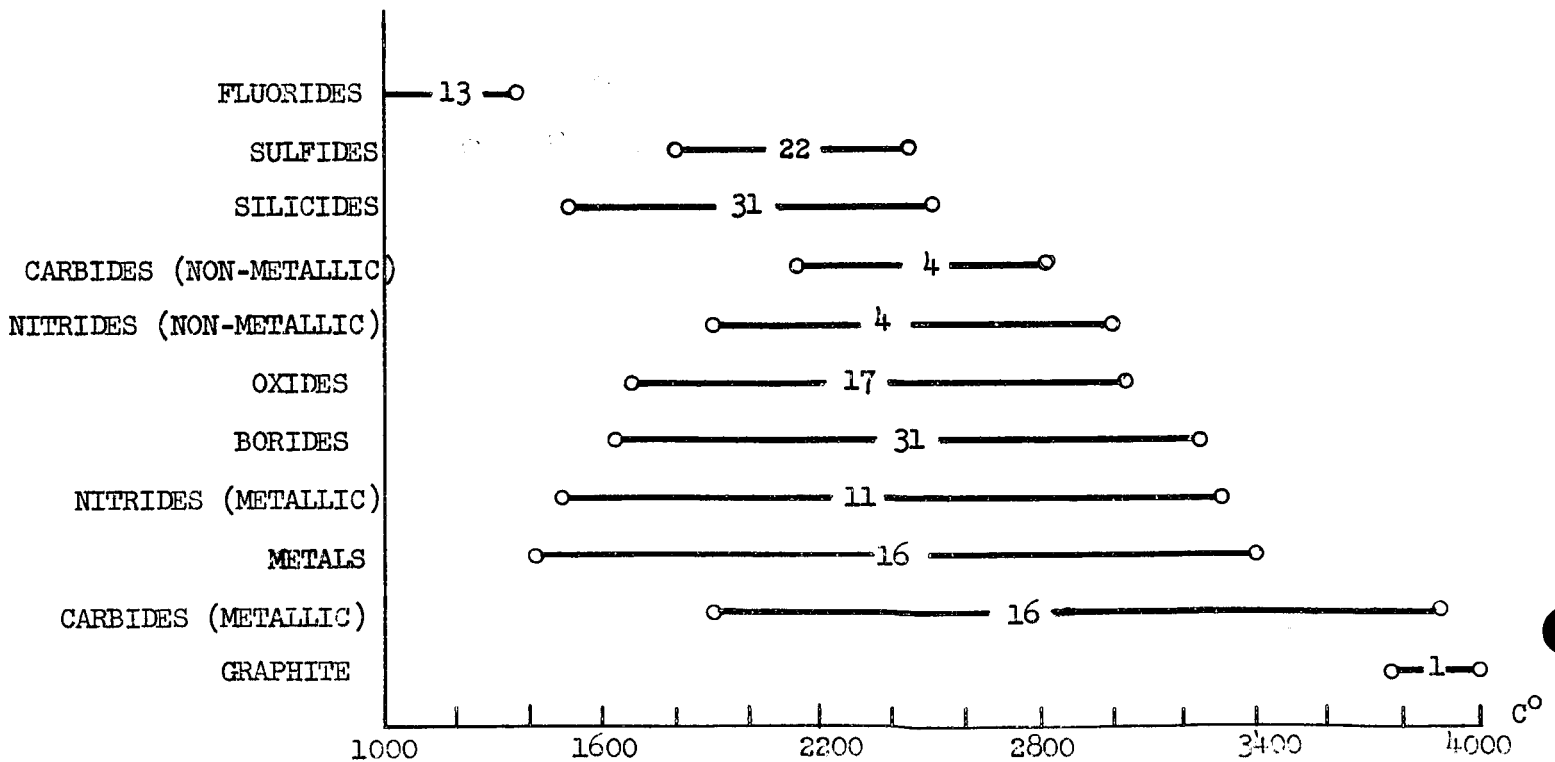Figure 4 - MELTING POINT OF CARBIDES

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Figure 5 - RANGE OF MELTING POINTS
OF REFRACTORY MATERIALS

Another informative compilation is shown in Table 2 in which various chemical elements are arranged in order of increasing thermal neutron capture cross-sections. Obviously for reactors (thermal) in which non-fissioning neutron capture is to be minimized, those materials with the higher capture cross-sections are to be avoided despite their high melting points (tungsten, tantalum and rhenium, for example).

Although melting point and nuclear cross-section are obvious factors in the selection of proper matrix materials for thermal reactors (cross-section can be relaxed considerably in fast reactor application), material compatibility, thermal conductivity, strength, fabricability and resistance to radiation damage are also important characteristics.

2.3 Cladding

Fuel element cladding is usually used to provide protection for the fuel and matrix material from corrosive attack by the reactor coolant and to contain fission products within the fuel element. It may also be employed to impart the required strength to the fuel element assembly. In view of these three additional features, the factors to be employed for selecting cladding material will differ to some extent from those which apply to the fuel and matrix materials. The operating temperature of the cladding will closely approach that of the enclosed fuel and matrix, and hence the melting point considerations mentioned above will apply. Also the nuclear cross-section concept will be important since the cladding is so intimately associated with the fuel. A close match of the thermal expansion of the matrix and cladding is an additional consideration.

It is possible that the cladding material will be the same as the matrix material as was the case in the Aircraft Nuclear Propulsion Program, Nichrome matrix material - Nichrome clad fuel elements. In this same program the proposed ceramic fuel elements contained UO_2 dispersed in BeO and clad with Al_2O_3 or other oxides to provide the necessary protection for the BeO with respect to moisture hydrolysis. In this latter instance the matrix material was actually the moderator, a technique which, when it can be employed, results in reduced reactor size. It is also worth noting that this concept forms the distinction between homogeneous and heterogeneous reactors; for when fuel and moderator are intimately mixed the term homogeneous applies, whereas when the fuel elements and moderator represent separate components, the construction is heterogeneous.

Following our discussion of the three segments of a fuel element, we can now appreciate the factors to be considered in selecting materials for fuel element application. A good fuel element must:

1) have a small affinity for capture of neutrons (of
    less importance when considering fast reactors),

TABLE 2

Macroscopic Thermal-Neutron-Cross Section (Capture) of the Elements

ELEMENT	CROSS SECTION (CM⁻¹)	MELTING POINT (°F)
CARBON (N)	0.0004	6700
BISMUTH	0.0008	520
BERYLLIUM	0.0012	2340
MAGNESIUM	0.0025	1202
LEAD	0.006	621
SILICON	0.007	2605
ZIRCONIUM (OK)	0.008	3450
RUBIDIUM	0.008	102
PHOSPHORUS (N)	0.009	111
CALCIUM	0.010	1560
SODIUM	0.012	208
ALUMINUM	0.013	1220
SULFUR (N)	0.018	246
BARIUM	0.020	1300
STRONTIUM	0.021	1420
CERIUM	0.021	1100
TIN	0.022	449
POTASSIUM	0.027	145
YTTRIUM (R)	0.036	2700
COLUMBIUM (OK)	0.060	4380
ZINC	0.070	787
GERMANIUM (N)	0.10	1760
THALLIUM	0.12	572
TELLURIUM (N)	0.13	840
GALLIUM (R)	0.14	86
IODINE (N)	0.14	237
MOLYBDENUM (OK)	0.15	4760
FE-CR-AL ALLOYS (OK)	0.18	2750
URANIUM (F)	0.18	2065
RUTHENIUM (R)	0.18	4500
ANTIMONY (N)	0.18	1166
ARSENIC (N)	0.19	1497
IRON (OK)	0.20	2800
THORIUM (F)	0.21	3300
CHROMIUM (?)	0.23	3540
LANTHANUM	0.24	1519
CESIUM	0.25	82
310 SS	0.25	2550
PRASEODYMIUM (R)	0.30	1700
COPPER	0.30	1981
TITANIUM (OK)	0.33	3300
VANADIUM (OK)	0.33	3150
NICHROME V (OK)	0.36	2550
SCANDIUM (R)	0.40	2190
NICKEL (OK)	0.41	2651
SELENIUM (N)	0.41	428
PALLADIUM (OK)	0.48	2829
PLATINUM	0.53	3224
MANGANESE	0.99	2275
OSMIUM (R)	1.1	4900
TUNGSTEN	1.2	6170
TANTALUM	1.2	5425
TERBIUM (R)	1.4	621
HOLMIUM (R)	2.0	
COBALT	3.0	2723
LITHIUM	3.2	367
SILVER	3.5	1760
LUTECIUM (R)	3.6	
THULIUM (R)	4.0	3100
HAFNIUM	4.6	5740
RHENIUM (R)	5.4	1945
GOLD	5.6	1313
INDIUM (R)	7.3	3571
RHODIUM (R)	11	
MERCURY	15	-38
IRIDIUM (R)	31	4449
DYSPROSIUM (R)	32	
EUROPIUM (R)	89	
BORON (N)	106	4200
CADMIUM	144	609
SAMARIUM (R)	308	2400
GADOLINIUM (R)	1100	

(F) FUEL
(R) VERY RARE
(N) NOT WORKABLE
(OK) POTENTIALLY USEFUL FOR FUEL ELEMENT APPLICATION

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2) have the required mechanical strength to withstand the imposed stresses due to drag loads of the coolant, vibratory loads due to moving coolant and the thermal stresses due to the thermal gradients,

3) have proper high temperature chemical stability (corrosion and compatibility characteristics),

4) have high thermal conductivity to minimize internal thermal gradients,

5) have ability to withstand radiation damage,

6) have ability to contain fission products,

7) be fabricable on a large scale basis.

It is all these requirements of a fuel element which make material selection an extremely difficult assignment.

The complexities involved in the fabrication of clad dispersion type fuel elements are not to be overlooked. The series of steps followed in the fabrication of the concentric ring fuel element design of the Aircraft Nuclear Propulsion Program is shown schematically in Figure 6. A close-up view of the fuel element is shown in Figure 7 and a completely assembled cartridge containing 18 fuel stages is shown in Figure 8.

A similar illustration of the fabrication process for coated ceramic BeO fuel tubes is given in Figure 9. Fuel tubes at various stages of the manufacturing process are shown in Figure 10.

3.0 MODERATORS AND REFLECTORS

Neutrons produced in the fission process by ejection from an atomic nucleus have very high energies of the order of 1 to 10 million electron volts (Mev). In terms of velocity the 10 million electron volts corresponds to about one-tenth that of light. Such high energy, fast moving neutrons (hence the name "Fast" neutrons) have a small probability of being captured by a fuel atom and in many reactors it becomes necessary to slow these neutrons down, i.e., decrease their energy, to promote more rapid and more frequent capture of the neutrons by the fuel atoms. This then leads to rapid propagation of the fission reaction and the generation of high power densities (in fast reactors based on the use of fast neutrons no such moderation is necessary but this type of reactor, at present, is less common).
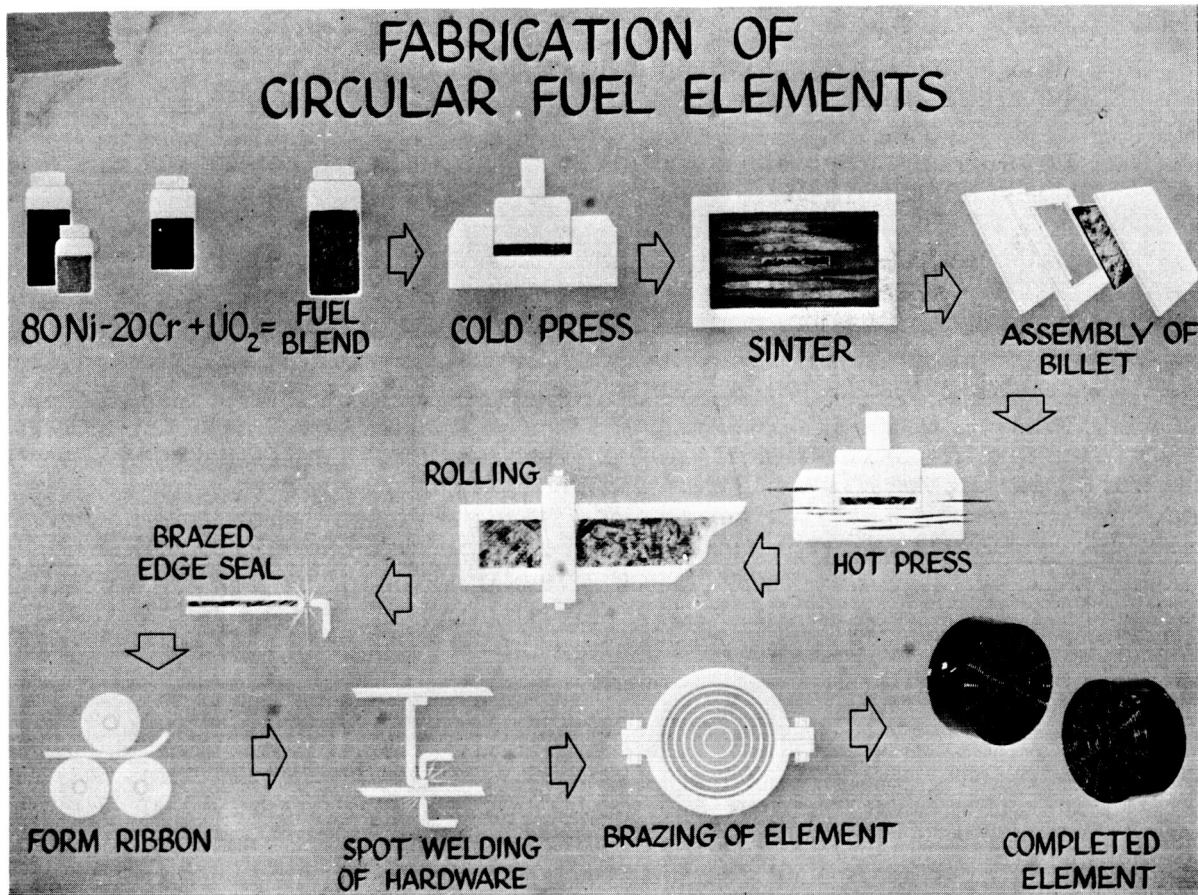
Fig. 6 - Fabrication procedure for nichrome-UO_2 dispersion type
fuel elements.
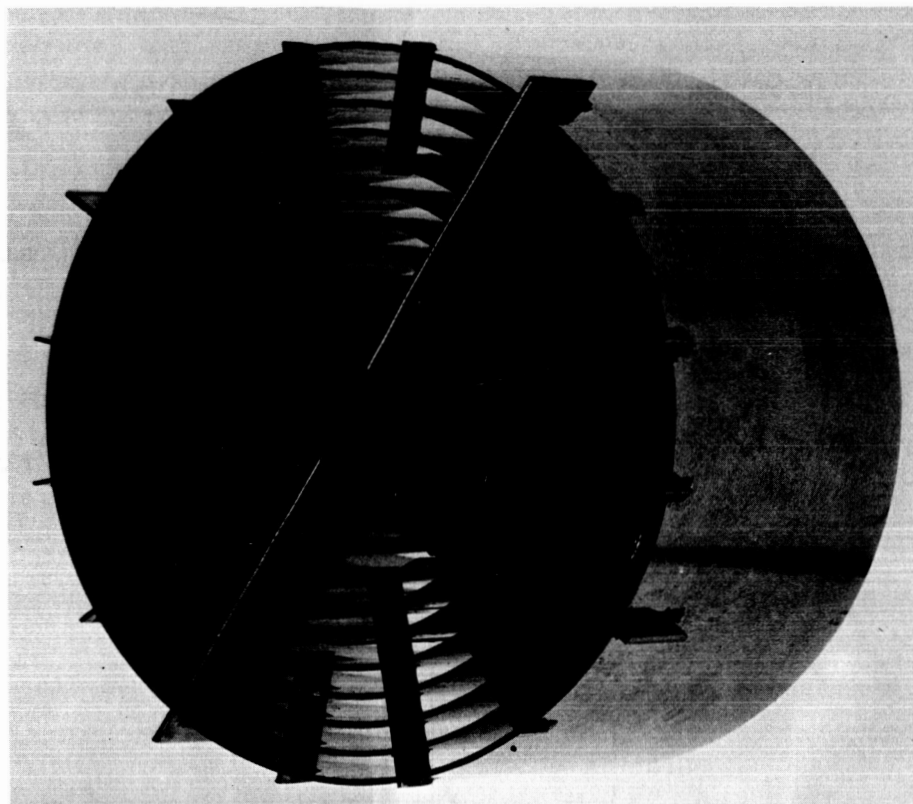
Fig. 7 - Concentric ring fuel element.
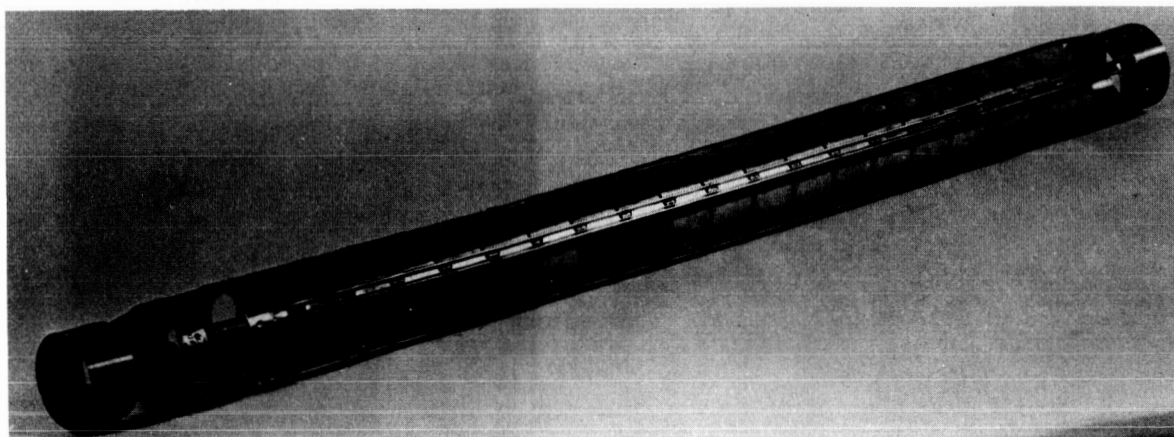
Fig. 8 - Complete fuel element cartridge.

Fuel   Stabilizer   Clad   Stabilizer
          | |
       Calcine   Calcine

          Blend   Plasticizers Ball Mill H₂O
                                         |
       React At   Mix
       3350°F |
                                      Pre-Extrude
Mechanically Reduce   BeO |
to -40 microns   Granulate

          Blend   H₂O   Plasticizers

                  Mix

              Pre-Extrude

               Granulate

                Co-Extrude

                   Dry

                Green Cut

                 Prefire

                 Sinter

            H₂ Treat at 2700°F

              Cut and Grind

               Inspection
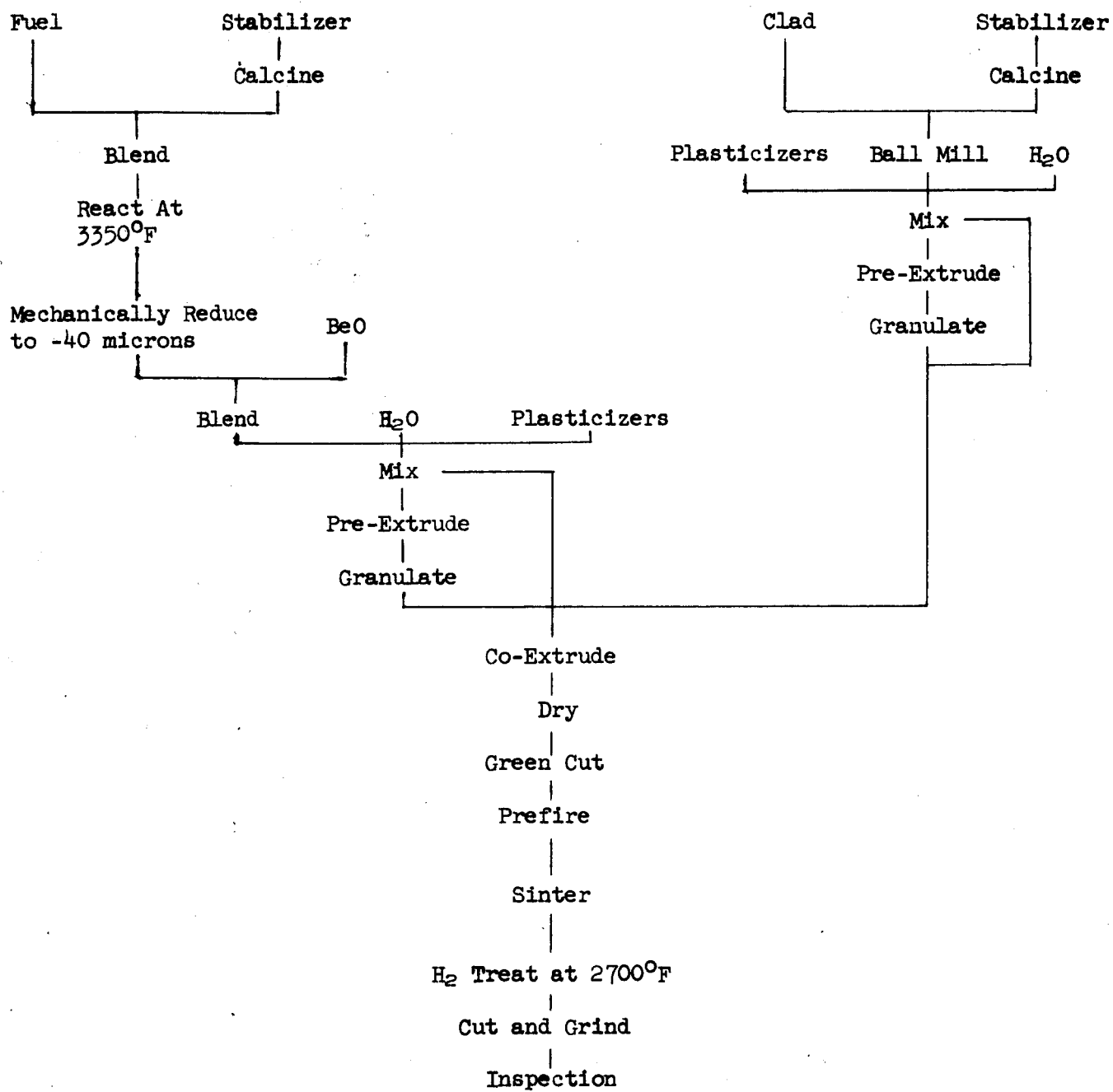
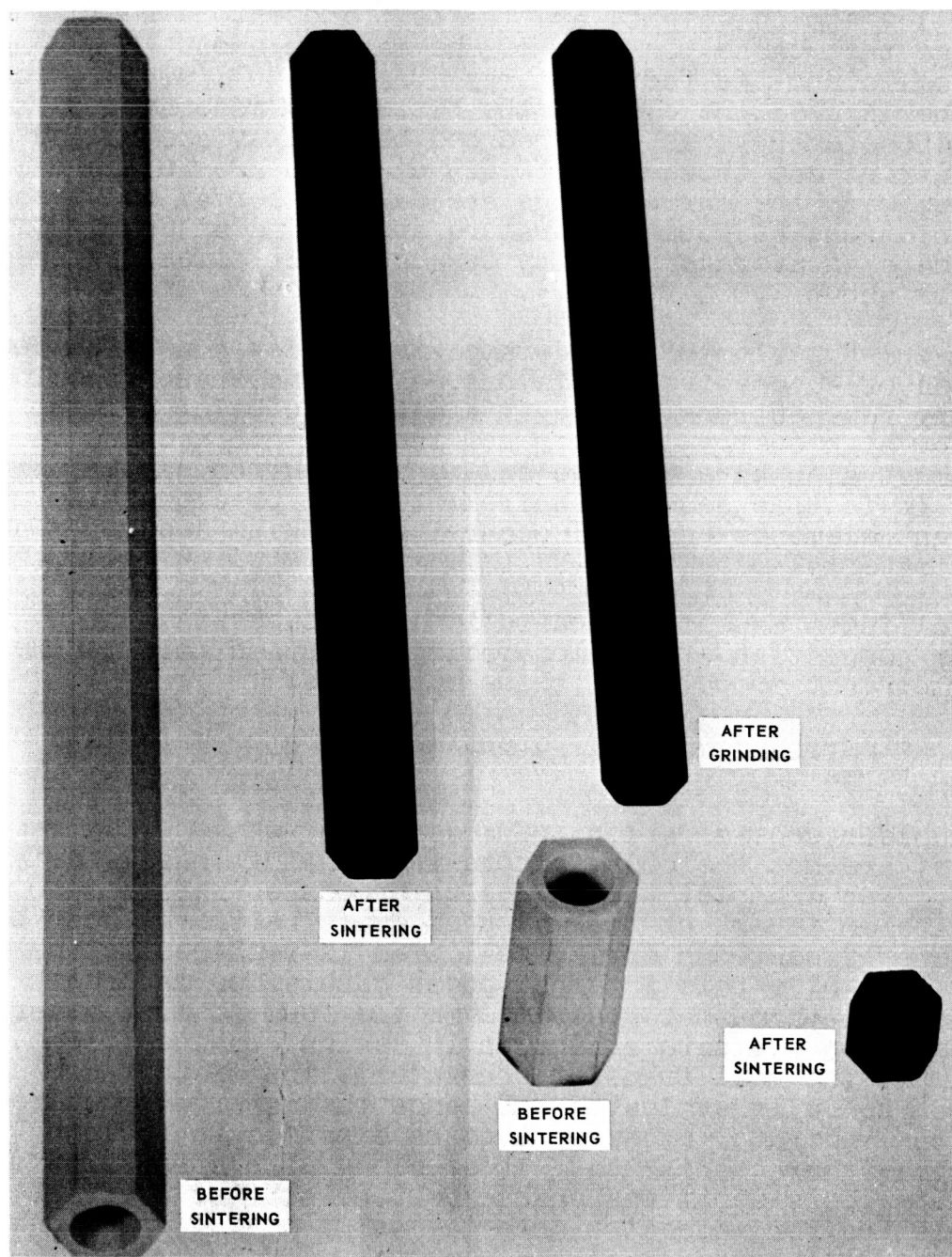Fig. 9 - Flow diagram for production of ceramic fuel elements.

Fig. 10 - BeO-UO$_2$ fuel elements at various stages of
manufacturing process.

The slowing down process is effected by the moderator and it does this by a scattering process (frequent collision of the neutrons with the atoms of the moderator). The light elements (those of low atomic weight) are the best moderators because on the average they cause the loss of the maximum amount of energy at each collision. As far as energy loss per collision is concerned, the lighter the element the better the moderator. The neutrons whose energy has been reduced below 1 ev are called "slow" neutrons and those below 0.025 ev which are in thermal equilibrium with the surroundings are called thermal neutrons (thermal neutrons at room temperature and having an energy of 0.025 ev have an average velocity of about ~~one~~ 7×10^{-6} times the speed of light or about 7,000 ft/sec.).

Since the function of a moderator is to slow a neutron down by a series of successive collisions, a good moderator should absorb very few neutrons. Thus in addition to being a light atom and causing high energy loss per collision with a neutron the material must also have what we call a low neutron capture cross-section. It is this second requirement which precludes the use of the light elements (natural occurring) lithium and boron as moderator materials.

The collision aspect of moderation also suggests a high density or population of the moderating atoms and this defines another important property of these materials.

A preliminary analysis of potential moderator materials can be made in terms of Table 3.

The nuclear considerations of moderators, while of extreme importance, are not the only factors influencing this materials selection. For example, the collision processes referred to above create an important thermal problem. The energy lost by the high energy neutrons during collision is retained by the moderator and while this internal power generation is orders of magnitudes less than that generated in the fuel it must nevertheless be given adequate attention in design considerations. Stated simply the heat transfer characteristics of this component must be fully evaluated and hence the material requirements of the moderator material must be expanded to include these thermal characteristics. These along with structural and fabricability considerations enable us to list the important characteristics for moderator material as:

1) low atomic number

2) low neutron absorption cross-sections

3) high density of the moderating element

TABLE 3

NUCLEAR CHARACTERISTICS OF POTENTIAL MODERATOR MATERIALS

Element or Compound	A, atomic or molecular weight	Density,[a] g/cm^3	N,[b] $\times 10^{-24}$	Scattering cross section σ_s epithermal, barns/atom	Absorption cross section σ_a, 0.025 ev, barns/atom or molecule	Macroscopic absorption cross section, $\Sigma_a = N\sigma_a$, cm^{-1}	Macroscopic scattering cross section, $\Sigma_s = N\sigma_s$, cm^{-1}	Logarithmic mean energy loss/collision,[c] ζ	Slowing-down power, $\zeta\Sigma_s$	Moderating ratio, $\sigma_s\zeta/\sigma_a$
Hydrogen, H	1.008	0.0090	0.0054	20.3	0.33	0.0018	0.11	1.000	0.11	61
Deuterium, D	2.02	0.0180	0.0054	3.3	0.00046	2.5×10^{-6}	0.018	0.725	0.013	5,200
Helium, He	4.00	0.0180	0.0027	0.8	-	-	0.0022	0.425	0.0009	∞
Beryllium, Be	9.01	1.85	0.124	6.1	0.009	0.0011	0.76	0.206	0.16	145
Carbon (graphite), C	12.0	1.60	0.080	4.7	0.0045	0.00036	0.38	0.158	0.060	165
Oxygen, O	16.0	0.014	0.0054	3.8	0.0002	1.1×10^{-6}	0.021	0.12	0.0025	230
Sodium, Na	23.0	0.97	0.0254	3.0	0.49	0.012	0.076	0.083	0.0063	0.53
Magnesium, Mg	24.3	1.74	0.043	3.4	0.059	0.0025	0.15	0.077	0.011	4.4
Aluminum, Al	27.0	2.70	0.060	1.35	0.215	0.013	0.081	0.071	0.0058	0.45
Beryllia, BeO	25.0	3.025	0.075	9.9	0.009	0.00066	0.72	0.173	0.12	190
Beryllium carbide, Be_2C	30.0	2.4	0.048	16.9	0.023	0.0011	0.81	0.193	0.16	145
Beryllium fluoride, BeF_2	47.0	1.986	0.025	15.4	0.029	0.00074	0.39	0.151	0.058	84
Light water, H_2O	18.0	1.00	0.033	44.4	0.66	0.022	1.47	0.925	1.36	62
Heavy water, D_2O	20.0	1.10	0.033	10.5	0.0011	36×10^{-6}	0.35	0.504	0.18	5,000
Sodium hydroxide, NaOH	40.0	2.1	0.032	27.1	0.82	0.026	0.87	0.77	0.67	26
Zirconium hydride, ZrH_2	93.2	5.61	0.036[d]	48.6	0.84	0.030	1.75	0.84	1.47	49
Biphenyl, $C_{12}H_{10}$	154.2	0.87	0.0034	259.4	2.54	0.00862	0.880	0.812	0.715	83
Polystyrene, $(CH)_n$	13.0	1.07	0.038[e]	25.0[g]	0.335[g]	0.013	0.95	0.842	0.80	62
Paraffin, $(CH_2)_n$	14.0	0.9	0.030[f]	45.3[h]	0.665[h]	0.020	1.36	0.913	1.24	62

[a] For the gases H_2, D_2, He, and O_2 an arbitrary density 100 times the density at NTP is assumed; i.e., a pressure on the order of 1,500 psi.

[b] Number of atoms or mole per cm^3.

[c] $\zeta = 1 - [(A - 1)^2/2A] \ln [(A + 1)/(A - 1)]$.

[d] Below 800°C.

[e] Number of (CH) units/cm^3.

[f] Number of (CH_2) units/cm^3.

[g] Per (CH) unit.

[h] Per (CH_2) unit.

TABLE 4

SOME TYPICAL REFLECTOR MATERIALS

		Be
I	Typical Thermal Reactor Reflectors	H_2O
		C
		BeO
		D_2O
- - - - - -	-	- - - -
		Al_2O_3
		ZrO_2
II	Typical Fast Reactor Reflectors	U^{238}
		Ni
		Cu
		Stainless Steel

Combinations of the I and II above can be used for thermal or epithermal reactor applications.

4) required temperature stability (melting point, vapor pressure, etc.)

5) required radiation stability

6) high thermal conductivity

7) adequate mechanical strength

8) adequate chemical stability or capable of being clad to obtain chemical inertness with respect to the environment

9) good fabricability

The function of a reflector is to scatter back or reflect as many neutrons as possible back into the active or fissioning portion of the reactor. Based on this function the nuclear requirements of the reflector are essentially the same as those of the moderator. It must have good scattering characteristics for reflection of the neutrons, and it must have a small cross-section for neutron capture. In fast reactors where neutron capture is of less concern, the major consideration then becomes the scattering cross-sections, and in this case high atomic number materials such as tungsten, molybdenum and even depleted uranium become important.

A list of some typical reflector materials is presented in Table 4.

An interesting sidelight is worthy of mention at this point for it emphasizes the type of significant technical accomplishment which is prompted by the newly created demands of high performance nuclear reactors. One of the most significant metallurgical achievements of the GE-ANPD direct-cycle program was the development of high temperature hydrogenous moderator materials. The size and weight of a thermal reactor can be decreased by using the moderator with the greatest slowing down power. Of all the nuclides, hydrogen, because its mass is about equal to that of the neutron, reduces neutrons from fission energy to thermal energies most rapidly. Thus, if the desirable features of small size and high power density are to be achieved, the use of a hydrogenous moderator with a relatively high hydrogen density is indicated. Analytical studies have shown that a hydrogen density of 4×10^{22} atoms per cubic centimeter or greater ($N_H = 4^*$) is necessary if the unique nuclear properties of hydrogen in the moderator are to be fully utilized.

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*N_H = number of hydrogen atoms per cubic centimeter of hydride $\times 10^{-22}$. This quantity can be calculated from the following expression:

$$N_H = \frac{(\rho MH_x)\ (\text{wt \% } H_2 \text{ in } MH_x)(N)}{A} \times 10^{-22},$$

where M is the metal, ρMH_x is the density of the hydride MH_x at 70°F, N is Avogadro's number, and A is the gram atomic weight of hydrogen.

In considering all the potential hydrides and hydroxides for use in a direct-cycle reactor, only zirconium and yttrium hydrides appeared attractive from a nuclear, metallurgical, and mechanical standpoint. These elements, as interstitial type of hydrides, are metallic in behavior and form.

Zirconium hydride was studied extensively because of the low cross-section of zirconium and the high theoretical N_H of the stoichiometric compound ZrH_2. Massive forms of reactor-grade zirconium were hydrided at 1400° to 1700^oF, with the resulting body remaining intact and with N_H varying from 4.0 to 6.6, depending on the temperature and pressure under which it was hydrided. The hydrided material was found to have a metallic character and a thermal conductivity equivalent to those of the parent metal.

The improved heat conductivity of the massive hydrided bodies, compared with that of hydrided bodies prepared by powder metallurgy, greatly alleviated the problem of removing moderator heat and permitted serious consideration of this type of material as a moderator in many reactor designs. Utilizing these techniques, hydrided zirconium was transplanted from a laboratory development to manufacturing production in the period 1957 to 1959. During this transition, cold- and hot-extrusion fabrication techniques were developed for zirconium which produced hollow hexagonal shapes 4.5 inches across the flats, 36 inches long, and 0.50 to 0.75 inch in wall thickness. Many hundreds of these shapes were extruded, hydrided, assembled, and tested in Heat Transfer Reactor Experiment No. 3 (HTRE No. 3). Studies on metallic bonded cladding to hollow hexagons were conducted but this technique was later abandoned. Advanced manufacturing studies on extrusion of zirconium materials also produced multiholed internally cooled biflute shapes and triflute shapes.

Recent laboratory development work on zirconium hydride technology demonstrated that with the use of certain grain-growth inhibitors it is possible to make sound massive zirconium hydride bodies to $N_H = 6.0$ and in some cases to $N_H = 6.6$ to 6.7. If gasified and at the same temperature and at 1 atmosphere of pressure, the hydrogen contained in 1 cubic centimeter of zirconium hydride (with $N_H = 6.6$) at 1400^oF would occupy 4.7 liters (a volume differential of 4700 to 1).

In an attempt to extend the operating temperatures of metallic hydrogenous moderators, it was determined that yttrium could be hydrided in massive form, in the same manner as zirconium, and that the thermal conductivity of yttrium hydride was considerably greater than the thermal conductivity of the parent metal. In 1956 the world's supply of yttrium was measured in grams, while the GE-ANPD requirements for a reactor would be in thousands of pounds. Consequently, a large-scale rare-earth-oxide chemical separation program was initiated in co-operation with the AEC. The basic problem was to obtain yttrium oxide of high purity,

particularly free of other rare-earth oxides (Gd, Sm, Eu, etc.), with high neutron absorption cross-sections. Through the combined efforts of Iowa State University, Dow Chemical Company, Michigan Chemical, the Atomic Energy Commission and GE-ANPD, a chemical separation process was developed to the extent that yttrium metal or oxide is commercially available today in thousand-pound quantities.

Exploratory work conducted on hydrided yttrium indicated clearly the superior high temperature hydrogen-retention properties of yttrium over all other materials tested. Zirconium showed a marked decrease in its hydrogen-retention properties above 1600°F, but yttrium showed a remarkable ability to retain hydrogen even when liquefied at 2400°F. At 2200°F the hydrogen-retention property of yttrium is nearly as great as that of zirconium at 1600°F. For a nuclear power plant in which the moderator temperatures will exceed 1600°F and reach 2000°F or higher, yttrium has a marked advantage over zirconium. Hydrided yttrium proved to be extremely brittle and therefore extremely sensitive to thermal stress and thermal gradients. Metallurgical resolution of this problem was attained by cladding the hydrided material with Fe-Cr-Al alloys and by alloying yttrium with zirconium and chromium. The hydrogen-retention properties of yttrium, Y-Zr, and Y-Cr alloys are compared in Figure 11 as a function of temperature. The advantage of obtaining the highest yttrium concentration is obvious.

A hexagonal shaped moderator for use in an ANP reactor assembly is shown in Figure 12. In assembly the full length fuel cartridge shown on a previous figure would be positioned within the center hole of this moderator section. This construction gives an excellent illustration of the heterogeneous reactor concept referred to above.

4.0 CONTROL ELEMENTS

Control elements serve to influence nuclear reactivity by the capture of neutrons and the farther the control element is inserted into the reactor core the more absorption occurs and the greater the decrease in reactivity. These materials, therefore, are called "nuclear poisons" and are characterized by their high cross-sections for neutron capture. One of the first materials to be used for reactor control was cadmium; but it has a very low temperature limitation, and for high temperature operation it can not be considered. Instead materials such as boron, hafnium, europium, gadolinium, dysprosium and samarium are of importance. These materials in elemental or compound form can provide the increased operating temperatures required in high power density reactors. For example, the four rare-earth oxides, europium oxide, gadolinium oxide, dysprosium oxide, and samarium oxide, are chemically stable in an oxidizing atmosphere to tempera-
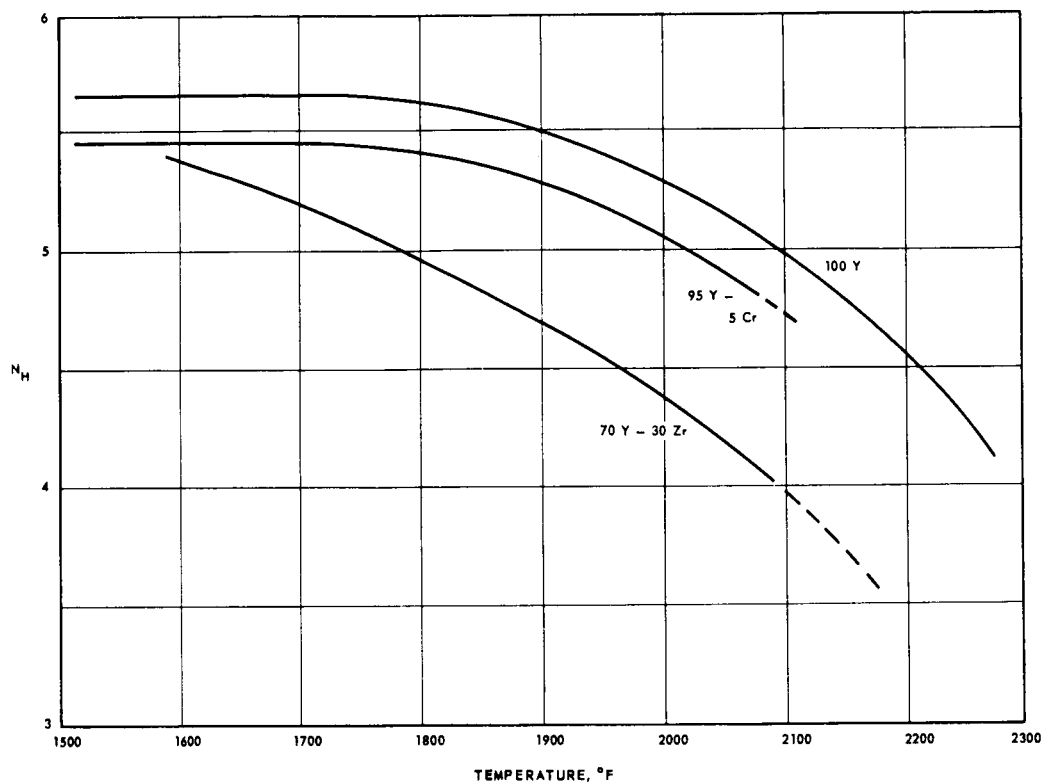
Fig. 11 - Isobars for hydrided yttrium and two yttrium alloys at 760 mm Hg.

Fig. 12 - Hydrided zirconium moderator.

tures of 1700°C and above. Long term tests at 1650° to 1750°C in a hydrogen atmosphere, at GE-NMPO, have shown that unclad europium and samarium oxides and refractory metal cermets containing these oxides must be clad to avoid reduction of the rare-earth oxides and subsequent loss due primarily to evaporation of the suboxide or metal. The rare-earth oxide cermets containing tungsten, rhenium, tantalum, and molybdenum show virtually no loss of the rare-earth oxide after hundreds of hours at 1650°C when clad with the molybdenum-50 rhenium alloy. Currently tungsten-25 rhenium alloy cladding is being developed, and it is expected that this will allow for even higher temperature operation.

It might be obvious that there is some similarity between control elements and fuel elements particularly in regard to matrix and cladding. For example, in the case of the rare-earth oxides a metal matrix is employed to increase the thermal conductivity of the control material, and it is clad to provide corrosion resistance and structural strength although the strength contribution made by employing the refractory metals as matrix materials is not to be overlooked. In view of the above and where applicable, material selection procedures for matrix and cladding material must be adopted, and in general these will closely follow those established for matrix and cladding of the fuel elements except, of course, that in the control element case the nuclear characteristics are completely different.

A list of some of the possible control element absorber materials is shown in Table 5.

Important factors governing the selection of control element materials are as follows:

1) high neutron capture cross-section

2) ability to operate in a high neutron flux for a long period of time without losing neutron capture effectiveness owing to the burnup of the absorbing isotopes

3) resistance to radiation damage effects (volume change, loss of corrosion resistance, loss of mechanical properties, etc.)

4) good mechanical strength

5) good thermal conductivity

6) good corrosion resistance

7) ease of fabrication

TABLE 5

SOME POSSIBLE CONTROL ELEMENT MATERIALS

Element	Atomic no.	Absorbing mass	Isotopes fraction	Thermal cross section, barns	Macroscopic thermal cross section, cm^{-1}	Absorbing reaction	Major resonance		Energy of other major resonances, ev	Neutron-absorbing capacity, g neutrons/cm^3
							Energy, ev	Cross section,[b] barns		
Lithium	3			71	3.29	n, alpha				0.006
		6	0.075	947						
Boron	5			755	106.7	n, alpha				0.044
		10	0.188	4010						
Cobalt[a]	27	59	1.00	37	3.37	n, gamma	132	7,000[c]		0.151
Rhodium	45	103	1.00	150	11.0	n, gamma	1.26	4,500		0.122
Silver	47			62	3.64	n, gamma				0.096
		107	0.519	30			16.6	630	42, 45, 52	
		109	0.481	84			5.12	12,500	31, 41, 57, 72, 86	
Cadmium	48	113	0.123	2550	118	n, gamma	0.178	7,200		0.009
Indium	49			190	7.26	n, gamma	1.46	30,000	3.86, 9.10	0.063
		113	0.042	58						
		115	0.956	197						
Cesium	55	133	1.00	29	0.25	n, gamma	5.9	2,600	22.6, 47.8	0.014
Samarium	62			5500	152	n, gamma				0.006
		149	0.138	50,000			0.096	16,000	0.86, 5	
		152	0.266	140			8.2	15,000		
Europium	63			4600	95	n, gamma				0.136
		151	0.478	9000			0.46	11,000	1.0, 3.4, 7.2, others	
		153	0.522	420			2.46	3,000	3.8, 9.0, others	
Gadolinium	64			46,000	1,400	n, gamma	2.58	1,000		0.015
		155	0.148	70,000						
		157	0.157	160,000						
Dysprosium	66			1,100	35	n, gamma	5.5	7,500	1.8, 16	
		164	0.282	2,600						
Erbium	68			166	5.48	n, gamma	0.46	2,600	6.2, 9.5; other	0.090
Hafnium	72			105	4.71	n, gamma				
		177	0.184	380			2.36	6,000		
		178	0.271	75			7.80	10,000		
		179	0.138	65			5.69	1,100		
		180	0.354	13			73.9	130		
Tantalum[a]	73	181	1.00	21.3	1.18	n, gamma	4.28	13,000	10.4, 14.0, 24.1, 39.3	0.090
Rhenium	75			84	5.57	n, gamma				0.110
		185	0.371	100			2.18	3,500		
		187	0.629	63			4.40	370		
Iridium[a]	77			430	30.1	n, gamma	5.6	5,400		0.116
		191	0.385	960			0.654	5,000		
		193	0.615	130			1.30	3,800		
Gold[a]	79	197	1.00	98	5.78	n, gamma	4.91	30,000	61.5	0.098
Mercury	80			380	15.45	n, gamma	23.3	150		0.014
		198	0.100	2,500			34.0	500		
		199	0.168							

[a] Considered too costly for use.

[b] The resonance cross section is frequently given for the isotope rather than the element, but the value refers to the natural element; the resonance cross section for the isotope would be found by dividing the value given by the abundance of the isotope.

[c] This resonance is 94 percent scattering.

In the case of fast reactors, control is usually provided by variation of the positions of the neutron reflector. This merely varies the number of neutrons which are scattered back into the core. Increased leakage obviously drives the reactor toward a subcritical state. In this case the material selection procedures follow those given for reflector materials.

5.0 SHIELDS

In the design of a nuclear reactor and its associated equipment, provision must be made for the attenuation of escaping nuclear radiations by some form of shielding. Not only is such shielding necessary for the protection of personnel, but a relatively high radiation background will interfere with satisfactory functioning of instruments used in various aspects of reactor operations and control. For mobile or space applications, the weight of the shield is of overriding importance.

The radiations which might escape from a reactor system include alpha and beta particles, gamma rays, neutrons of various energies, fission fragments and even protons resulting from the (n,p) reactions. As far as shield nuclear design is concerned, however, only gamma rays and neutrons need be considered since these are by far the most penetrating. Any material which attenuates these radiations to a sufficient extent will automatically reduce all the others to negligible proportions. Thus, the problem of shielding a reactor involves three aspects: (1) slowing down the fast neutrons, (2) capturing the slowed-down, or initially slow, neutrons, and (3) absorbing all forms of gamma radiation.

For slowing down fast neutrons, the elements of low mass number are the best materials, and these have been previously presented in Section 3.0 on moderators. The situation can be greatly improved by introducing an element (or elements) of fairly high mass number as they are able to reduce the energy of very fast neutrons by inelastic collisions.

The next aspect of reactor shielding to consider is that of capturing the slowed-down neutrons. This is a relatively simple matter as the probability of further slowing down and subsequent capture is large even though the shield does not contain an element of fairly large cross-section for the capture of slow neutrons. It should be noted that it is important that the capture of the neutron should not result in high energy gamma radiation.

The third and final aspect of the shielding problem is the absorption of the various gamma rays, both primary and secondary. The energy of these gamma rays is significant, and they are most effectively absorbed by material of high density. It is evident, therefore, that the shield constituent which serves as the inelastic scatterer of neutrons will also serve as the absorber of gamma radiation.

From the above considerations it is obvious that all materials can function as a shield material to a certain extent. Some of the common shield materials are listed in Table 6 which was also presented as Table 1 of Lecture 5 and 6, "Shield Physics". In this table the density and relaxation lengths for fast neutrons and gamma rays are given and clearly show the influence of density on the absorption of gamma rays. It is also apparent that combinations of elements are probably needed for compact reactors for space applications in which weight is of primary importance. Shield materials for compact reactors may contain high concentrations of hydrogen, lithium, beryllium, and boron atoms which serve as good neutron scatterers. Yttrium hydride has the highest hydrogen retention at elevated temperatures of any usable hydrogenous material known. The material has a room temperature hydrogen density comparable to water at room temperature (0.85/1) and is comparatively stable to temperatures of about 1200°-1300°C. Yttrium hydride has a further advantage in that the high atomic weight of yttrium contributes to gamma attenuation. For higher temperature application, beryllium oxide and tungsten and/or tantalum may be considered for neutron-gamma shields.

Another factor that is significant in selection of shield materials is related to temperature and environment surrounding the shield. Absorption of radiation releases heat in the shield and necessitates cooling and/or cladding. The cladding may serve two functions; one, the protection of the shield material, and two, to retain the components that comprise the shield. The problems are somewhat akin to the problems faced in fuel element and moderator development in that structural and compositional integrity need to be retained. To illustrate the diversity of size, shape, and method of fabrication that may be employed in shield production, see Figures 13, 14 and 15.

6.0 STRUCTURE

The two principal structural components of a reactor are the external pressure shell and the internal core-support structure. The pressure shell may be thermally insulated from the hot core and radiation, in which case the problems encountered in selecting structural materials are similar to those found in selecting materials for a heat exchanger operating at the same temperature, pressure, and heat transfer rate. However, if the pressure shell is to contribute to the shielding, energy is deposited directly in the shell by neutron and gamma absorption, causing internal heating and thermal stresses, and a shell cooling system may then be required. Obviously, the shell material strength - density ratio should be as high as possible to permit minimum-weight pressure shells.

TABLE 6

RELAXATION LENGTHS[*] FOR FAST NEUTRONS AND GAMMA RAYS

Material	Density, g/cm^3	Relaxation Length, cm	
		Fast Neutrons	Gamma Rays
Lithium Hydride	0.73	7.8	41
Water	1.0	10	30
Graphite	1.6	9	19
Beryllium	1.8	7.5	18
Beryllium Oxide	2.3	7.5	14
Concrete	2.3	11	15
Barytes Concrete	3.5	8	10
Iron	7.8	6	3.7
Lead	11.3	9	2.5
Uranium	18.7	5.6	1.4
Tungsten	19.4	5.7	1.5

[*]Based in part on the high energy radiation leaving a large reactor core.
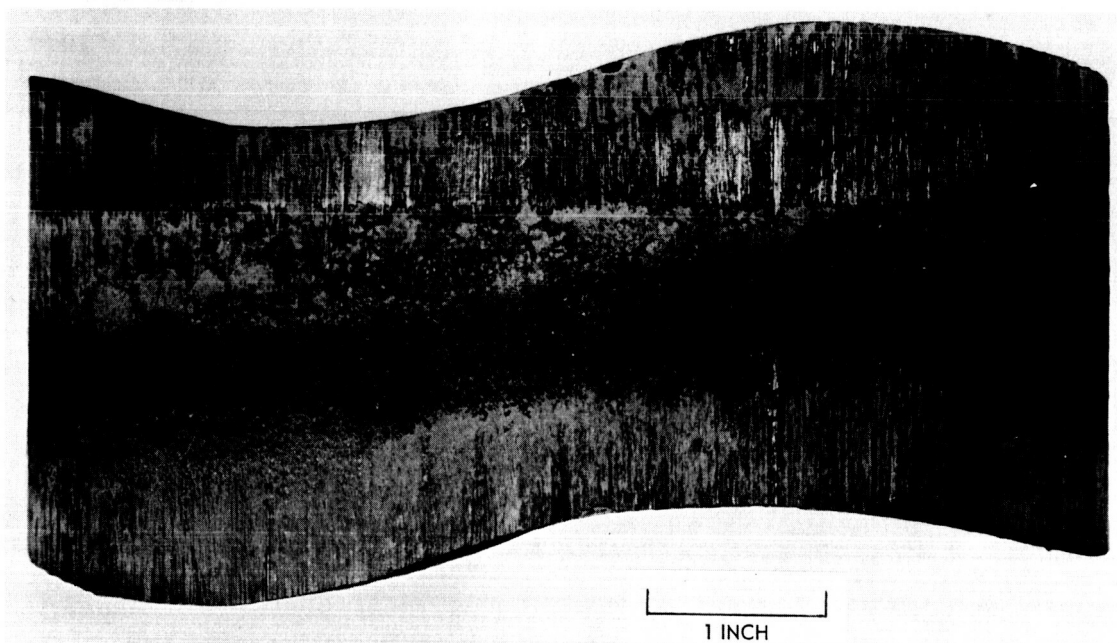
Fig. 13 - Hot-pressed beryllia and zirconium boride block

A 12- BY 18- BY 20-INCH PIECE OF PERFORATED
STAINLESS STEEL HONEYCOMB (0.001-INCH FOIL
WITH 1-INCH SQUARE CELLS PERFORATED BY A
1/2-INCH HOLE)
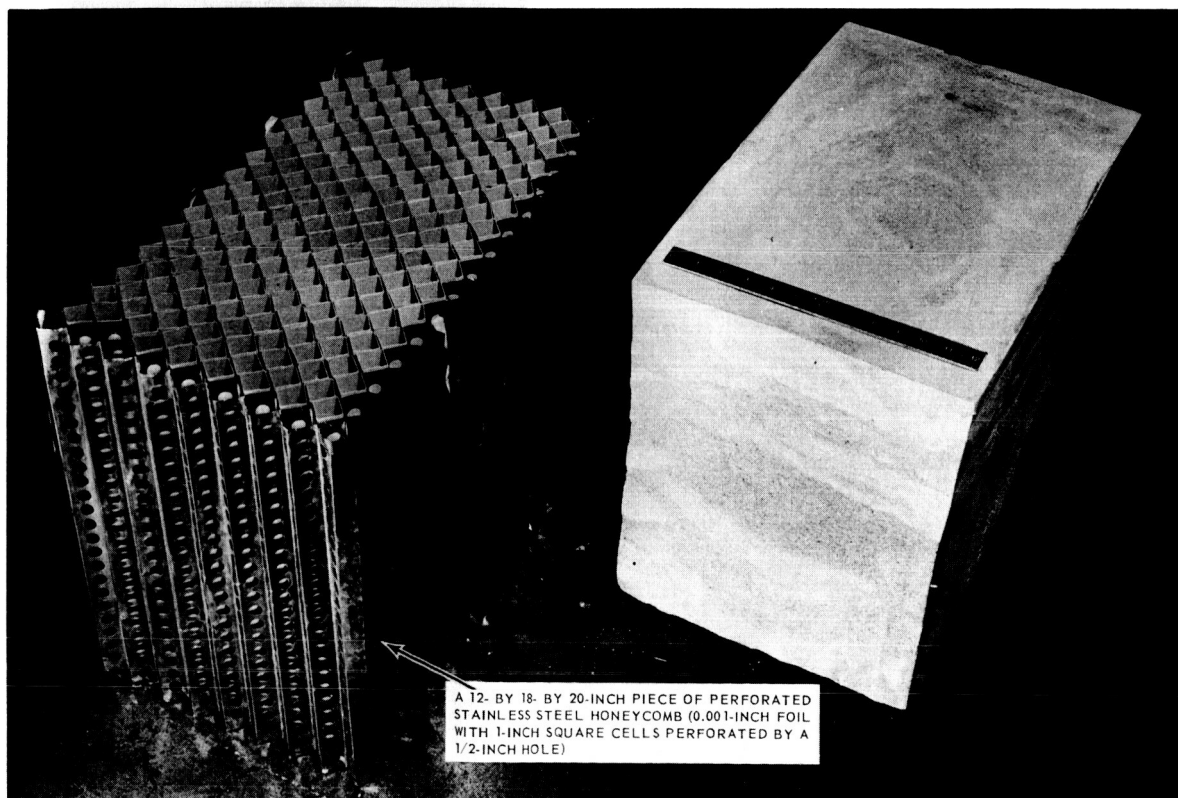
Fig. 14 - Block of cold pressed lithium hydride and metal matrix
which may be incorporated in the lithium hydride to im-
prove the strength and thermal shock resistance.
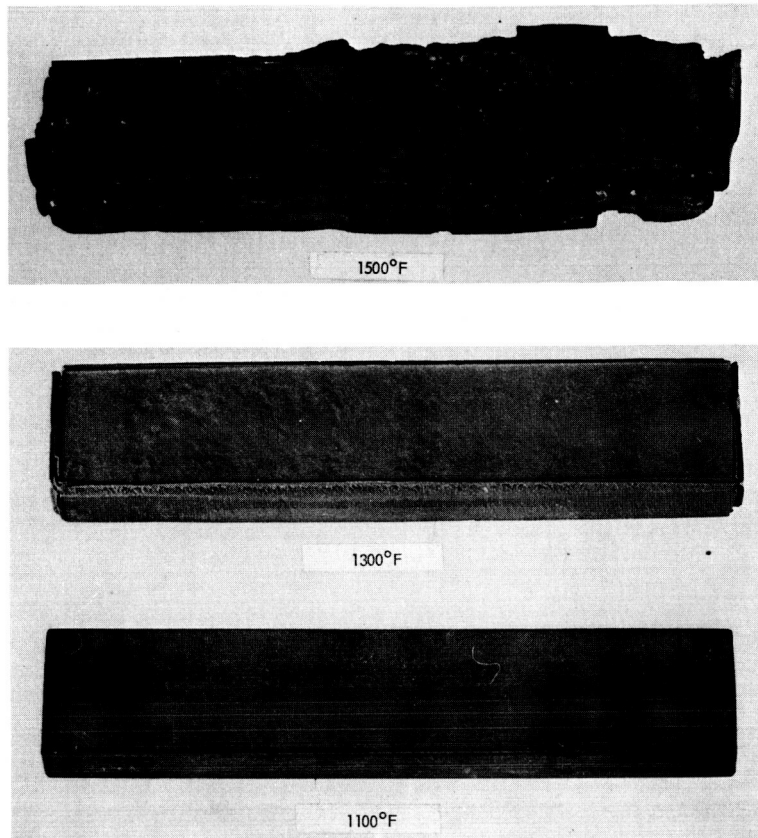
Fig. 15 - 90W-6Ni-4Cu alloy prepared by powder metallurgy
technique after 100 hour oxidation test at various
temperatures.

The internal core-support presents the added problems of exposure to high coolant temperature and nuclear radiation. In thermal reactors the conservation of neutrons has been of overriding importance and has resulted in primary considerations of low cross-section materials, aluminum, magnesium, and zirconium.

As the operating temperature increases, the corrosion and strength become increasingly severe, and higher cross-section materials, such as stainless steel, nickel-base alloys, niobium, and molybdenum, may be considered at the cost of fuel addition.

In fast reactors, while the absorption cross-section is not a major problem, the core is usually small and the power density high; so heat transfer characteristics, thermal stresses, and thermal shock resistance are of primary importance.

The selection of a structural material is usually based on a compromise to obtain the best combination of nuclear, physical and mechanical properties, corrosion resistance, stability, and fabricability.

The following comments should serve to illustrate some of the physical, mechanical, and nuclear properties applicable to some of the structural materials.

Magnesium - Well established technology - good strength to weight ratio - low thermal neutron absorption cross-section - corrosion resistance and low melting point preclude use at elevated temperature.

Molybdenum - Refractory metal - excellent strength with good ductility above 1000°C - thermal neutron absorption low in comparison to other refractory metals - oxidizes readily at temperatures above 500°C - easily fabricated.

Nickel - Tough - moderately strong - good corrosion resistance - oxidation resistant to 1000°C - neutron absorption higher than austenitic stainless steels.

Tantalum - High neutron absorption cross-section - extremely refractory - high affinity for oxygen and nitrogen gases at elevated temperatures and for hydrogen at low and intermediate temperatures - fabrication and machinability excellent - high cost.

Titanium - Good combination of mechanical strength and corrosion resistance - stable against oxidation up to 700°C - extremely ductile - easily fabricated - neutron absorption higher than stainless steels.

Tungsten - Same as for tantalum except that fabrication is more difficult and compatibility with hydrogen is excellent.

Zirconium - Good corrosion properties, comparable to titanium - lower neutron absorption cross-section than titanium - oxidation resistance poor above 300°C.

Aluminum - Well established technology - excellent formability - large supply - low cost - good nuclear properties - excellent heat transfer - corrosion resistance and strength good at low temperatures - low melting point and strength limit its usefulness to low temperatures.

7.0   REACTOR COOLANTS

In many cases the selection of the reactor coolant is just as involved as the selection of materials of construction for the various reactor components. In these cases nuclear heat transfer and fluid dynamic considerations usually prevail in this selection; although, obviously such factors as availability, cost, corrosiveness, and ease of handling are also important. It is also possible to identify certain instances wherein the coolant is specified, and no flexibility whatsoever is allowed in coolant selection. For instance, in the direct-cycle reactor for aircraft propulsion, air cooling is imposed and must be accepted. Similarly, in a reactor for nuclear rocket application, thrust-to-weight considerations, in terms of specific impulse, dictate the use of hydrogen; and although some compromise may be made in this instance, the spectrum of potential coolants is drastically narrowed.

Whether the coolant is selected or imposed, it determines the operating environment for many, and in some cases all, of the reactor components. For this reason all material selection procedures must await coolant identification, for as mentioned previously, materials which might perform perfectly in one environment might be totally inadequate in another. In addition to the primary considerations of corrosiveness of the atmosphere with respect to specific materials, an item of equal importance concerns the effect of atmosphere on material properties. This latter subject has received only limited attention; but as higher and higher reactor operating temperatures are pursued and required, it becomes one of our major considerations.


8.0   PROPERTY MEASUREMENTS

Based on the foregoing discussion it has been seen that the selection of materials for use in reactor components is determined in large measure by the degree to which these materials are possessed of certain properties. It also should be recognized by now that the different components have different property

requirements but that in each case we can specify the exact nature of the desired characteristics. We should quickly acknowledge at this time though that while writing the property specifications may be a fairly simple task we should not interpret this as an admission that numerical values for all these properties are available. Indeed property measurement represents one of the many important phases of materials technology, and it is well to point out that at this moment the property measurements necessary to the accurate assessment of the true potential of the highest melting point materials for reactor application are the subject of many research programs in numerous laboratories throughout the country.

Some appreciation of the magnitude of the property measurement program may be gained by a brief reference to a list of the specific properties usually required in a materials evaluation program. Such a list is shown below. Now a knowledge of all these properties is not necessary to the design of every reactor, but this listing will serve to indicate the scope of interest for a materials engineer engaged in the evaluation of materials for nuclear reactor operation.

Tensile Properties - The tensile properties are obtained from a test in which the specimen is loaded in tension until rupture occurs. During this test the loads and corresponding strain (elongation) are recorded to yield a stress-strain curve. A typical stress-strain curve is shown below.
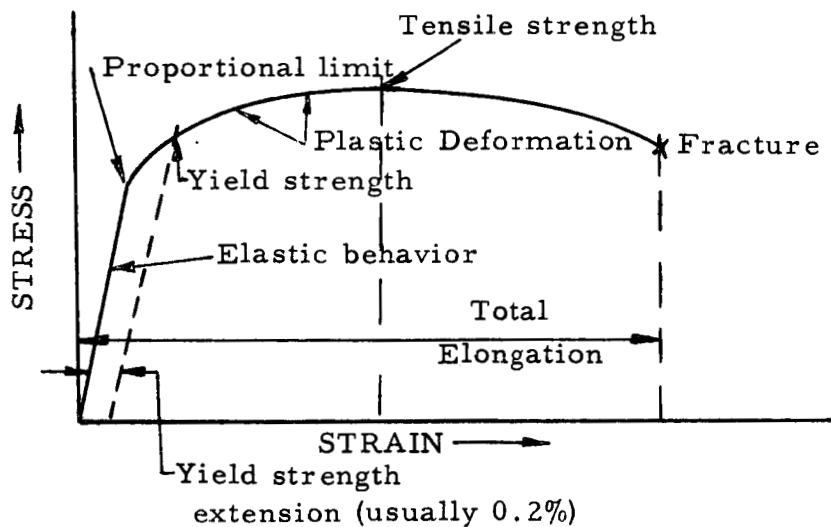
From such a test the following tensile properties can be obtained:

a) <u>Modulus of Elasticity</u> - The modulus of elasticity is the slope of the linear portion of the curve and is obtained by dividing the strain increment into the stress increment.

b) <u>Proportional Limit</u> - The proportional limit represents the departure from the straight line relationship.

c) <u>Yield Strength</u> - Of much greater practical importance is the yield strength, which is defined as the stress at which some predetermined amount of permanent strain occurs, usually 0.2%.

d) <u>Tensile Strength</u> - The tensile strength is the maximum stress that the material sustains in the tensile test. The deformation up to this maximum stress is essentially uniform. Beyond the point of maximum load, continued pulling of the specimen results in local deformation until fracture occurs.

e) <u>Poisson's Ratio</u> - The ratio of the transverse contraction to the longitudinal elongation of a specimen strained below the elastic limit; essentially a statement of constancy of volume during deformation.

t = Rupture time
C = Total elongation
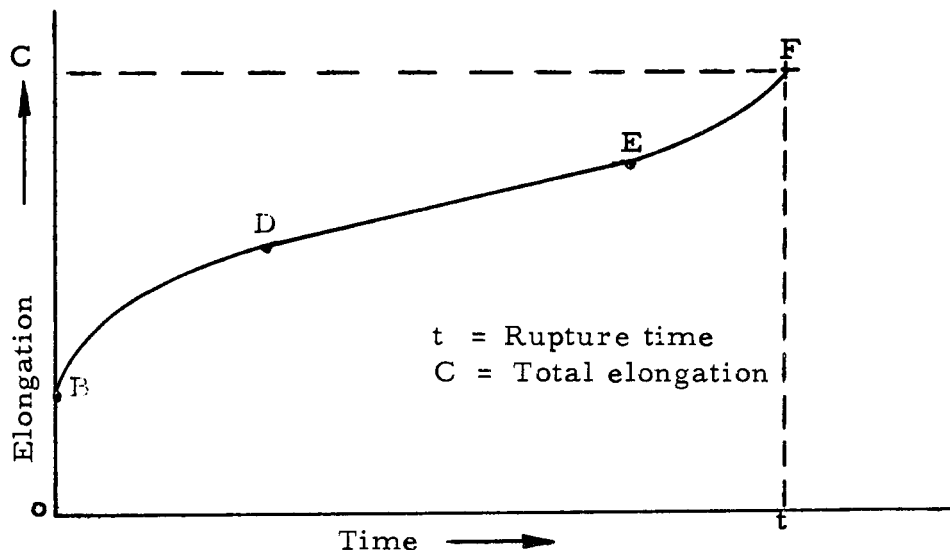
O B   is the instantaneous elongation on application of load.

B D   is the first stage of creep, deformation is rapid, decreasing to a steady rate as a function of time.

D E   is the time during which creep continues at a steady rate; this is the second stage of creep.

E F,   the third stage, is a period of accelerating deformation with time until rupture occurs.

f) <u>Stress-Rupture</u> - Stress-rupture is the time necessary to fracture under a certain stress. This property is normally determined where large deformations are permissible, the limiting feature being fracture.

g) <u>Fatigue Properties</u> - The fatigue test is used to study the resistance of metals to failure under conditions of many cycles of reversed stressing. The number of cycles necessary for failure is recorded at various stress levels.

h) <u>Reduction in Area and Elongation</u> - A measurement of the reduction in area and elongation (after fracture) are a measure of ductility.

i) <u>Compression Properties</u> - Compression tests are the opposite of tension tests, and properties can be obtained which are similar to those in tension: modulus of elasticity, proportional limit, yield strength, etc.

j) <u>Hardness</u> - Hardness is a measure of the resistance of a material to indentation by an indenter of fixed geometry under a static load. A useful relationship exists between hardness and tensile strength for a number of metals, and thus hardness testing may serve as an excellent tool for inspection and control work.

k) <u>Creep</u> - Creep is a term applied to the gradual plastic deformation of a metal, usually at elevated temperatures, when it is subjected to a relatively low stress. Creep of metals occurs in three stages as indicated above. However a creep test need not be continued beyond the second stage, since the purpose of the creep test is to determine creep rate.

l) <u>Fatigue Strength</u> - is the maximum stress which the material will withstand without failure during a specified number of cycles of stress.

m) <u>Endurance Limit</u> - is the stress below which no failure is expected in a million cycles or more.

n) <u>Thermal Expansion</u> - Different materials expand different amounts when heated and contract in the same way when cooled. Solid objects expand a certain fraction of their length for each degree rise in temperature. Thus each material has a <u>coefficient of linear expansion</u> which gives the fractional change <u>in length</u> for each degree change in temperature.

o) <u>Thermal Conductivity</u> - All substances do not transfer heat by conduction at the same rate. The amount of heat which can be conducted from one place to another in a given time is measured by conductivity. It is the time rate of transfer of heat by conduction, through unit thickness, across unit area for unit difference of temperature.

p) <u>Density</u> - Mass per unit volume of material; usually expressed as grams per cubic centimeter or pounds per cubic foot.

q) <u>Melting Point</u> - Temperature required to cause solid to liquid phase transformation.

r) <u>Vapor Pressure</u> - Equilibrium pressure exerted by the material at a given temperature.

s) <u>Heat Capacity</u> - Amount of heat required to raise the temperature of a unit mass of material by one degree; usually expressed as BTU/pound-$^\circ$F or calories/gram-$^\circ$C.

t) <u>Compatibility</u> - Refers to the ability of materials to be in contact without interacting.

u) <u>Chemical Stability</u> - Refers to all phases of chemical behavior and may even include chemical compatibility covering resistance to attack by other materials or environments and resistance to dissociation on heating.

v) <u>Fabricability</u> - Refers to the ability of a material to be rolled, formed, machined, drilled, molded or otherwise formed into desired geometry.

w) <u>Radiation Damage</u> - Extent to which material nature is altered upon exposure to nuclear radiation; includes changes in physical dimensions and all property values.

x) <u>Fission Product Retention</u> - Ability to retain within the material structure the fission products produced in the nuclear fission process.

y) <u>Thermal Stress Resistance</u> - Ability to resist without fracture the stresses induced by operating temperature gradients; pertins to steady state temperature gradients as well as transient gradients during temperature cycling.

z) <u>Modulus of Rupture</u> - Nominal stress at fracture in a bend test or torsion test; in bending the modulus of rupture is the bending moment at fracture divided by the section modulus.

aa) <u>Nuclear Cross-Section</u> - Defines the probability of the reaction of the material with a certain type of radiation (neutron or gamma ray); includes capture, elastic and inelastic scattering, transport, activation and resonance, etc.

bb) <u>General Nuclear Characteristics</u> - Refers to number of isotopes formed, half-life of these isotopes, decay scheme of radioactive isotopes, beta and gamma energy released in decay process, etc.

APPENDIX A

As a further guide to one who may be required to become more familiar with reactor materials, the following references should be reviewed.

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Volume 1, Materials
Interscience Publishers
1962

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1955

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Neutron Absorber Materials For Reactor Control
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Naval Reactors, Division of Reactor Development
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High Temperature Materials
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Reactor Shielding Design Manual
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TID-7004, U.S. Government Printing Office, Washington, D.C.
March 1956

Reactor Materials Properties, S.J. Paprocki and
R. F. Dickerson, Nucleonics, Vol. 18, No. 11, November, 1960

Aerospace Engineering, January 1963, High Temperature
Issue, Vol. 22., No. 1.

APPENDIX B

Glossary of Nuclear Terms

Absorber - A material which has a high affinity for neutrons, absorbing them without fission. (See "Control Rod" and "Shielding".)

Alpha ray - (alpha particle) A positively-charged particle consisting of two neutrons and two protons.

Atomic mass - Generally the mass of an atom in atomic mass units, based on the mass of an oxygen atom which is valued at 16. Atomic mass is approximately equal to the total of protons and neutrons in the nucleus of an atom.

Atomic number - The number of protons (positive charges) in the nucleus of an element, each element having a characteristic number.

Barn - A unit area (10^{-24} cm^2) for describing nuclear cross sections.

Beta ray - Negatively - or positively-charged electrons emitted by certain radioactive materials.

Burn up - (% depletion) The percentage of a fissionable material used as fuel which is destroyed by fission or capture, compared to the original quantity of that fissionable material present.

Capture cross-section, neutron - A number applied to a nucleus describing its relative affinity for absorbing neutrons without fissioning. The higher the cross-section, the greater the affinity for neutrons.

Chain reaction - In nuclear physics, this term refers to the fission of an atom which causes fission of other nuclei which, in turn, cause more fission, etc.

Cladding - This term refers to a thin layer, usually of metallic material, which covers a solid fuel element in a reactor to prevent corrosion in the presence of the coolant, and to prevent escape of fission products. Sometimes called coating.

Closed cycle - A method of operation which utilizes a material repeatedly, such as in the cooling system of a nuclear reactor, in which the coolant is repeatedly cycled through the reactor core, the heat exchangers, and back to the reactor core, without removal from the system.

Control rod - A rod made of or containing materials of high capture cross section (usually boron or cadmium) which controls the neutron flux in a reactor. Movements of the control rods in the reactor core change the reactivity of a reactor and can make a reactor subcritical, critical, or supercritical.

Coolant - A liquid or gas for cooling reactor parts, particularly the fuel; the medium by which the heat generated within a reactor is drawn from the reactor.

Core - Refers usually to the fuel assembly or fuel-moderator assembly in a nuclear reactor; the section of a reactor containing the fissionable material.

Critical - The state of a reactor when a self-sustaining chain reaction has just been achieved (the zero-power state).

Critical experiment - The study of the neutron behavior in a reactor assembly to determine critical mass, control rod effectiveness, etc. The studies are made at and below criticality (see "Critical").

Critical mass - The minimum mass of fissionable material in a nuclear reactor necessary to achieve a self-sustaining chain reaction.

Deuterium - An isotope of hydrogen, with a nucleus of one neutron and one proton; known as "heavy hydrogen".

Elastic collision - Collision of particles with no loss of kinetic energy, resulting in a mutual deflection of the particles.

Elastic scattering - Scattering of particles by elastic collision, i.e., without any loss of kinetic energy.

Electron - A negatively-charged, subatomic particle of mass equal to 9.107×10^{-28} gms. The outer shell of an atom comprises electrons spinning about the atomic nucleus.

Enriched material - Nuclear fuel containing more than normal content of a radio-active isotope. Usually refers to uranium containing more than the 0.7% of U^{235} isotope present in natural uranium.

Fast neutrons - Neutrons resulting from fission that have lost relatively little of their energy; energies exceeding approximately 0.1 Mev.

Fission fragments - The nuclei of lighter elements resulting from the fission of a heavier element.

Fission poisons - Fission fragments which absorb neutrons unproductively; especially used for those having high absorption cross-sections.

Fission products - This term refers to the isotopes of elements, usually radio-active, ranging in atomic number between approximately 34 to 58, resulting from the fission of a heavy element such as uranium or plutonium.

Fuel - Fissionable material used in nuclear reactors for the production of energy.

Fuel element - Nuclear fuel in a particular shape or configuration, usually clad to protect the active portion from corrosion, and to prevent fission product escape.

Gamma ray - A short-wave radiation emitted by some radioactive atoms; energies usually ranging between 10 Kev and 10 Mev.

Half-life - The length of time required for the radioactivity of a substance to decrease to half of its original activity.

Heavy water - Water in which the ordinary hydrogen is replaced by its heavy isotope, deuterium; used as a moderating material.

Inelastic collision - Collision of particles in which there is net kinetic energy change due to internal energy changes.

Inelastic scattering - Dispersion of particles in various directions (scattering) caused by inelastic collision.

Intermediate neutrons - Kinetic energies greater than that of thermal neutrons but less than that of fast neutrons, i.e., energies of approximately 1 to 10^5 ev.

Irradiation - The exposure to radiation.

Moderator - A material of low atomic mass and low capture cross section, used for slowing down (moderating) the speed of neutrons (by collisions).

Neutron - An elementary particle with mass number 1, but having no electrical charge.

Nuclear reactor - Essentially an arrangement of fissionable material, coolant, moderator, and shield, designed for maintaining a nuclear chain reaction.

Open cycle - A method of operation which utilizes a material only once, such as in the cooling system of a nuclear reactor in which the coolant passes through the core of the reactor only once and is then discarded.

Poison - Any material which absorbs neutrons unproductively; especially those having high absorption cross-sections.

Proton - An elementary, positively charged particle of mass number 1.

Radiation damage - Deleterious changes in the structure and physical properties of materials resulting from nuclear radiation.

Reflector - A material surrounding the core of a nuclear reactor, and designed to reflect neutrons, that otherwise would escape, back into the core.

Safety rod - A special control rod of a nuclear reactor used to stop the chain reaction rapidly in case of emergency.

Shield - A mass of dense radiation-absorbing material surrounding the core of a reactor, making approach to the reactor relatively safe.

Shim rod - A control rod used for start-up and shut-down of a reactor, and controlling large amounts of reactivity.

Slow neutrons - (thermal neutrons) Neutrons which have lost most of their fission energy because of impact with atoms in water or other moderators, and have energies below about 1 ev.

Thermal cross-section - The capture cross-section of a nucleus of thermal or slow neutrons.

Thermal neutrons - (slow neutrons) - Neutrons having kinetic energies of about 0.025 ev.

Thermal shield - An inner shield designed to absorb escaping thermal neutrons and other radiation and thus to prevent undue heating of the outer shield. The shield may be specially cooled.

Uranium (U) - Element of atomic number 92, metallic in form. Several isotopes of uranium are fissionable, and hence can serve as nuclear fuels.

Uranium-235 (U^{235}) - A fissionable isotope of uranium; the only naturally occurring nuclear fuel.

Appendix C

Some typical design data for various materials are shown in this section. The effect of temperature on these properties is noticed to be quite pronounced in many cases. The data for graphite are presented to illustrate a case in which strength increases with increasing temperature.
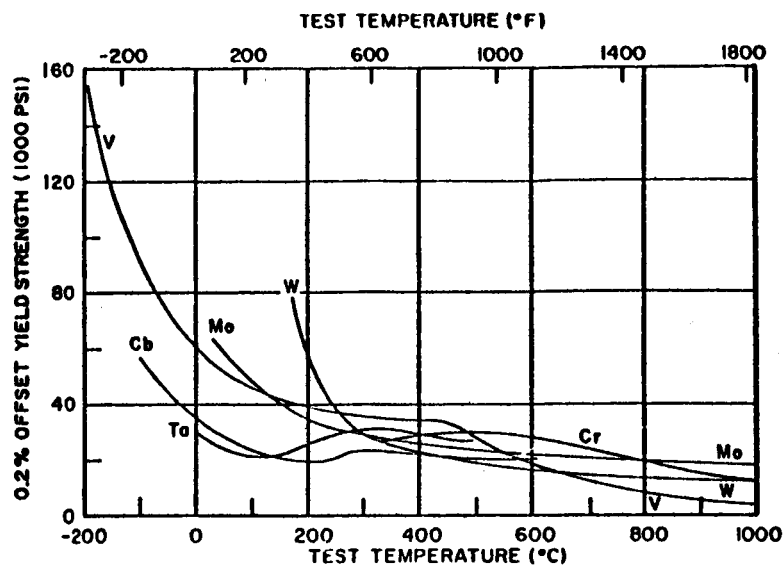
Fig. 19

YIELD STRENGTH OF SIX REFRACTORY METALS FROM -200 TO 1000°C
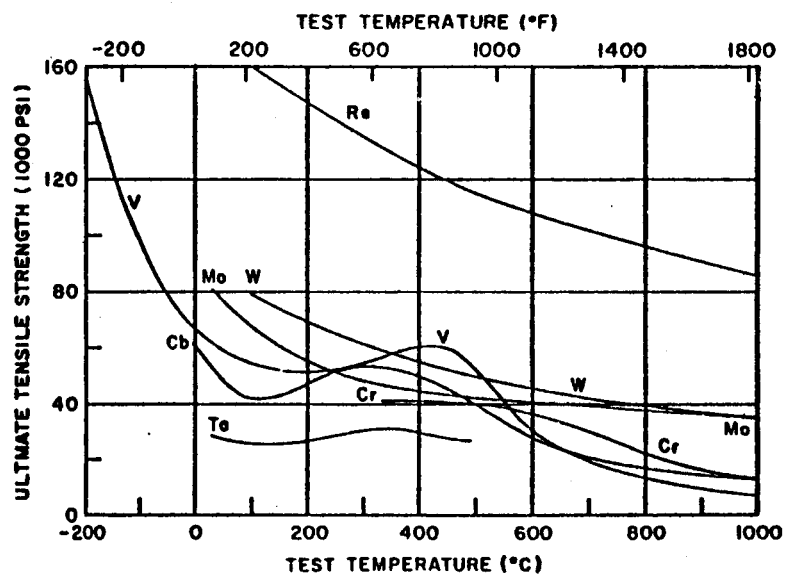
Fig. 16

ULTIMATE TENSILE STRENGTH OF SEVEN REFRACTORY METALS
FROM -200 TO 1000°C
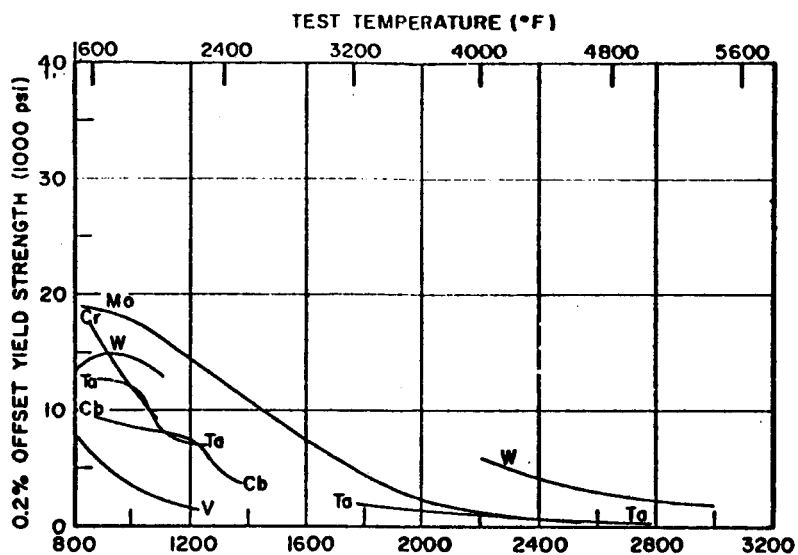
Fig. 20

YIELD STRENGTH OF SIX REFRACTORY METALS FROM 800 TO 3000°C
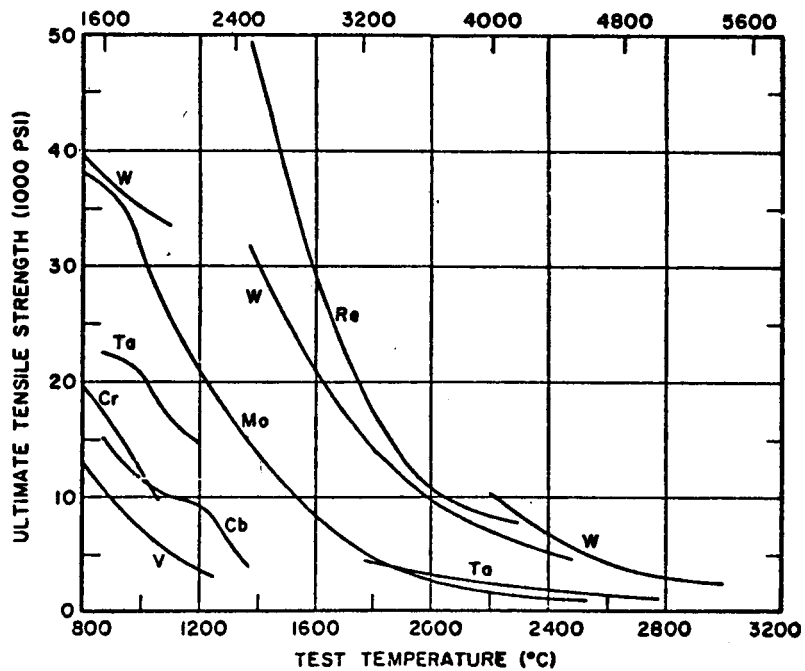
Fig. 17

ULTIMATE TENSILE STRENGTH OF SEVEN REFRACTORY METALS FROM
800 TO 3000°C

46
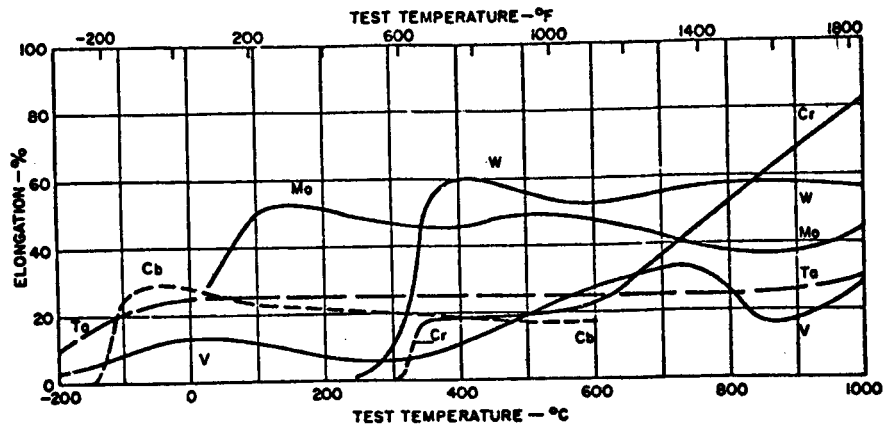
Fig. 21

ELONGATION VERSUS TEST TEMPERATURE FOR SIX REFRACTORY
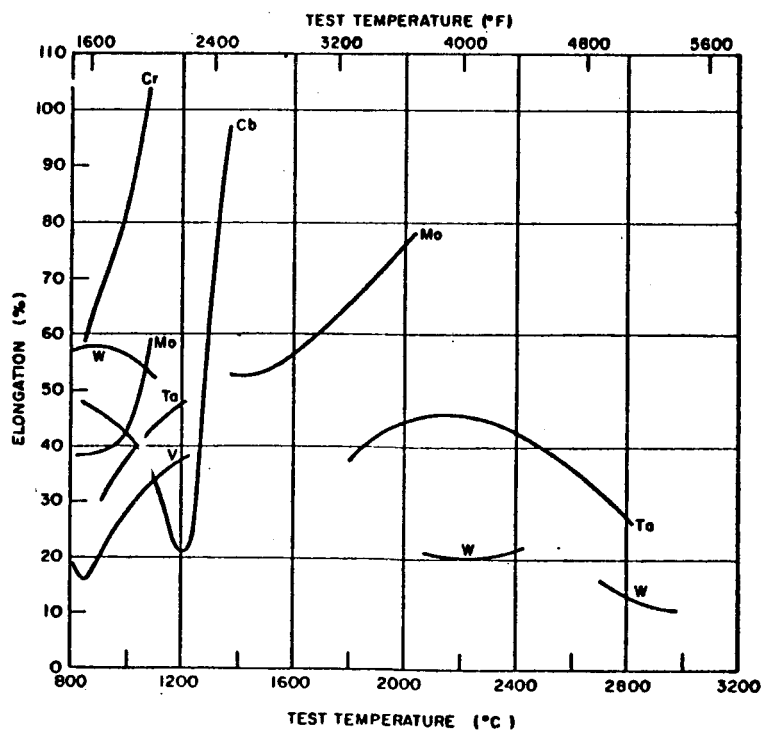METALS FROM -200 to 1000°C

Fig. 22
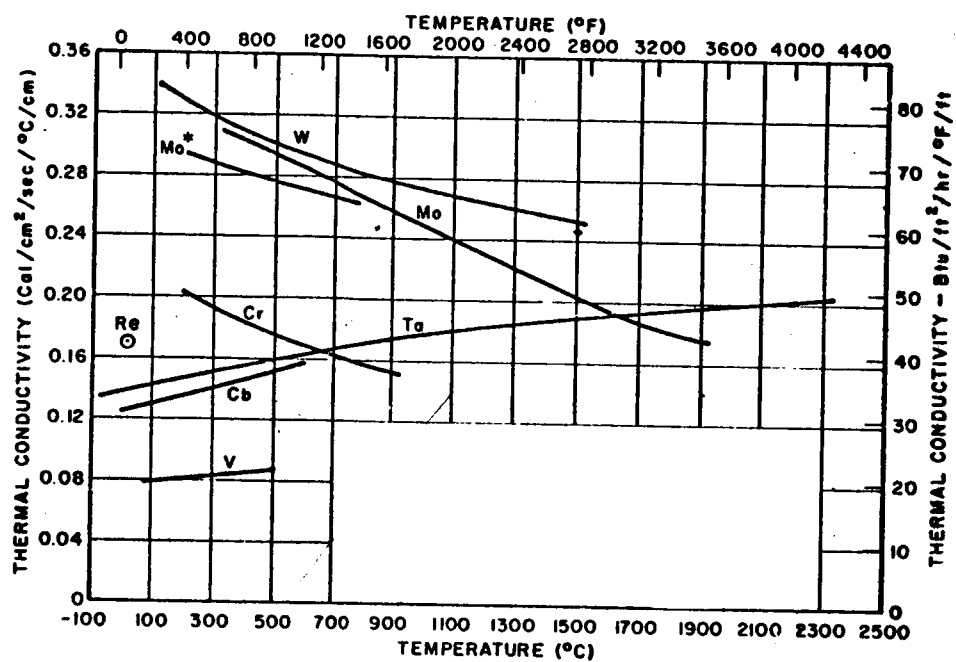
ELONGATION OF SIX REFRACTORY METALS FROM 800 TO 3000 °C

Fig. 23

THERMAL CONDUCTIVITY AS A FUNCTION OF TEMPERATURE
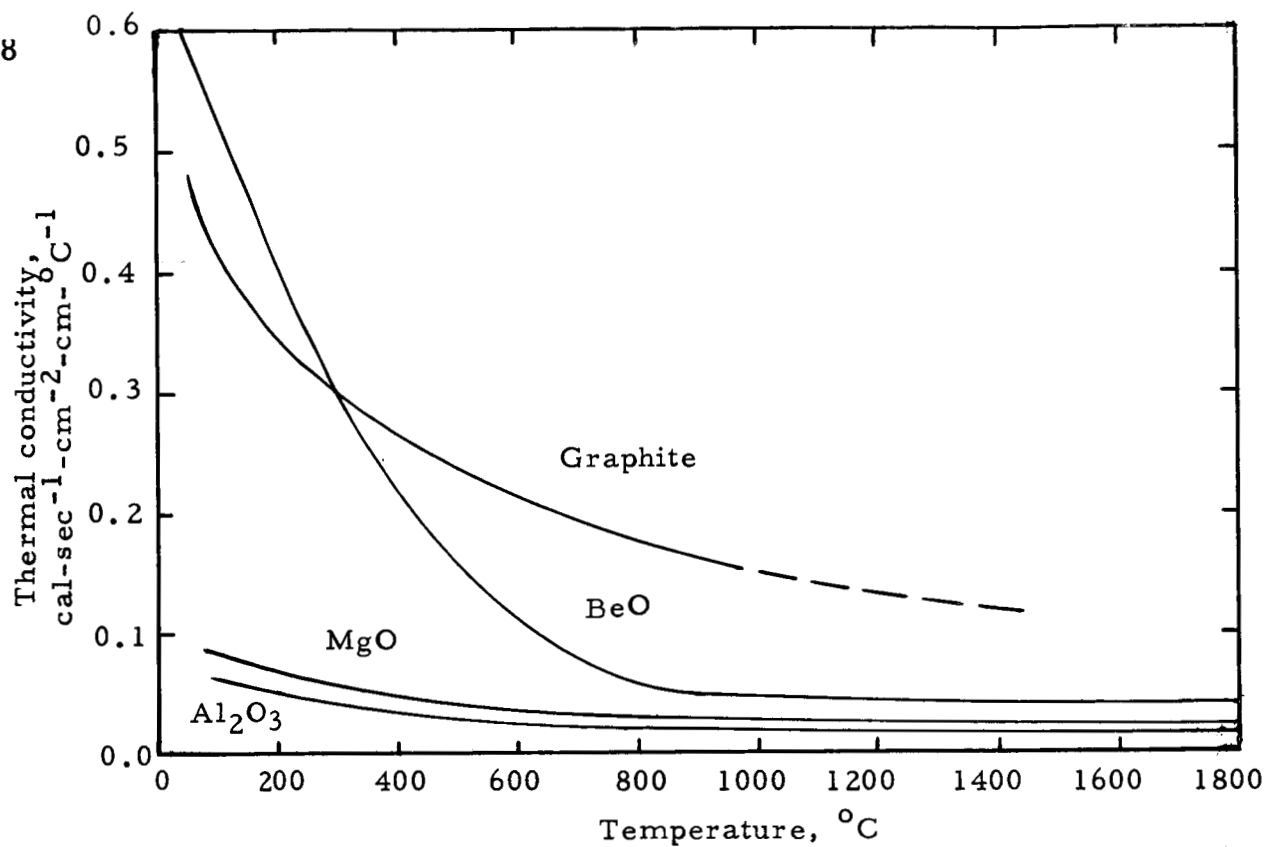
Fig.24 - Thermal conductivity of polycrystalline oxides
corrected to theoretical density.
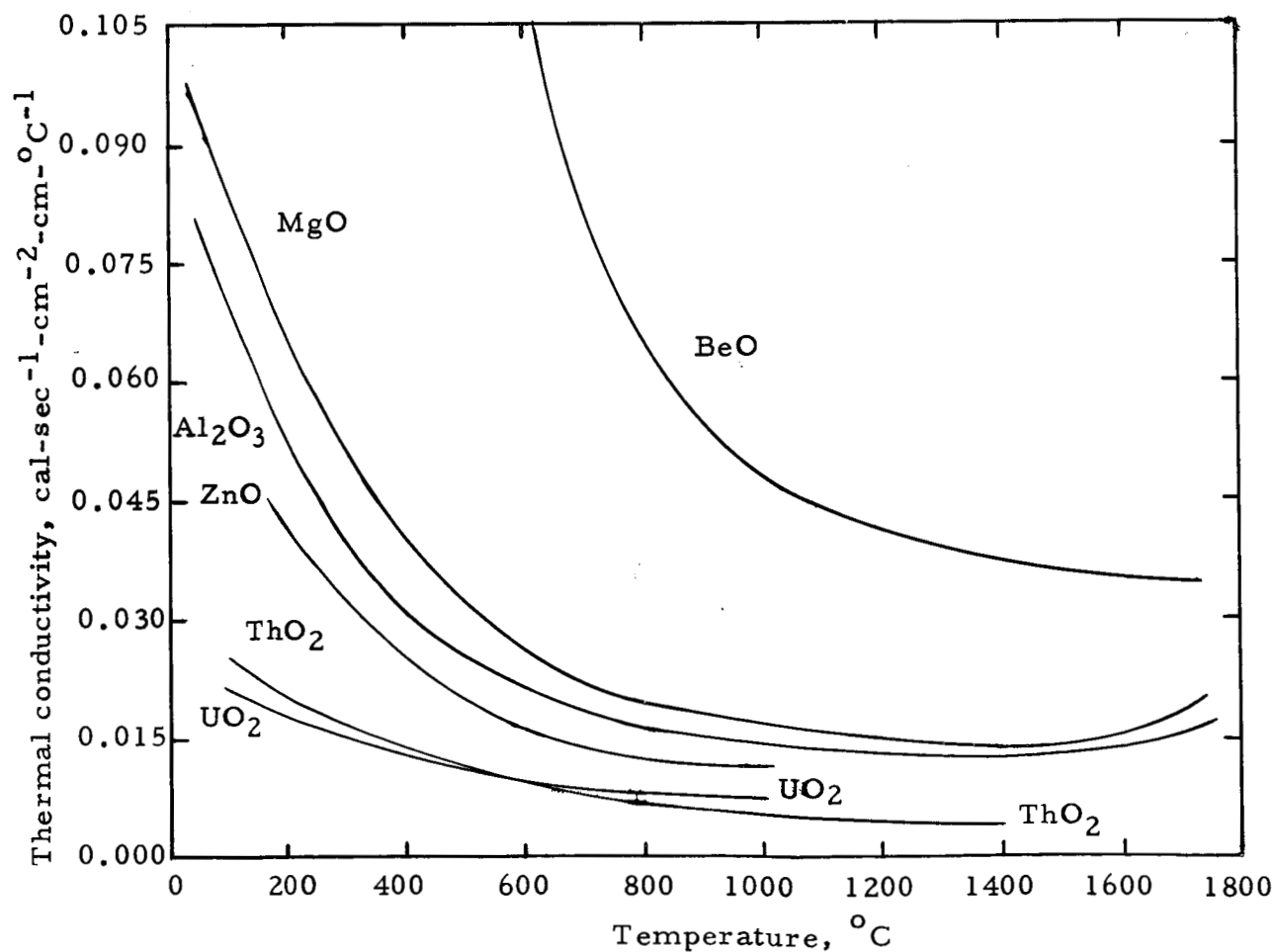
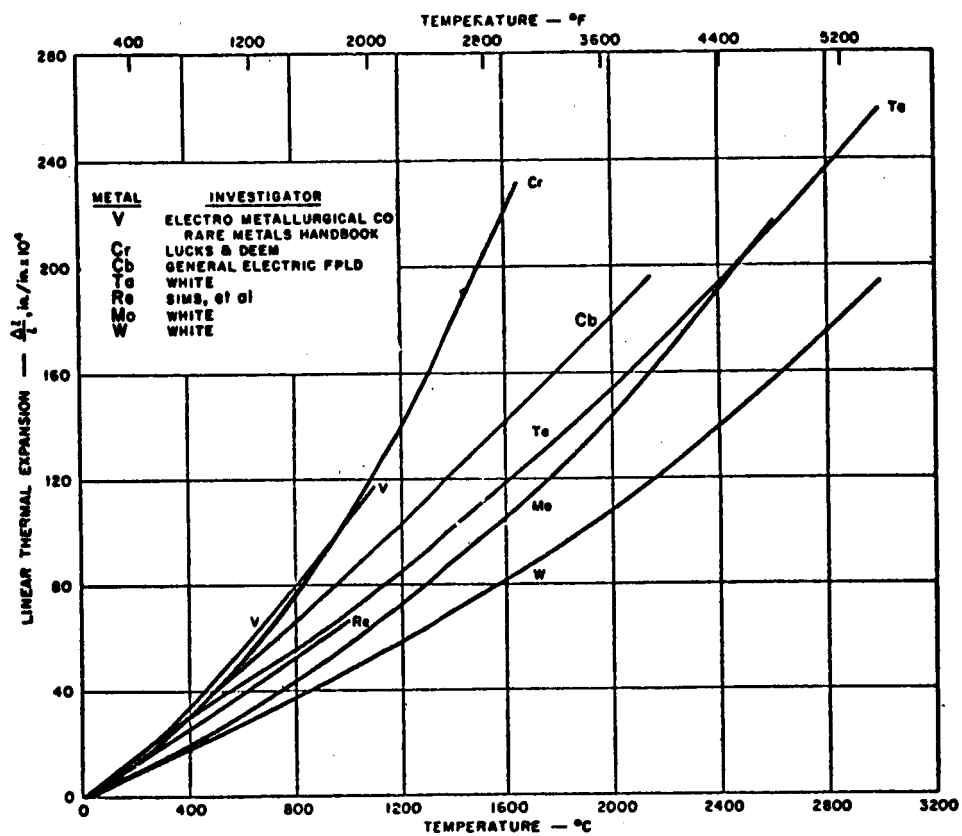Fig.25 - Thermal conductivity of polycrystalline oxides corrected
to theoretical density.

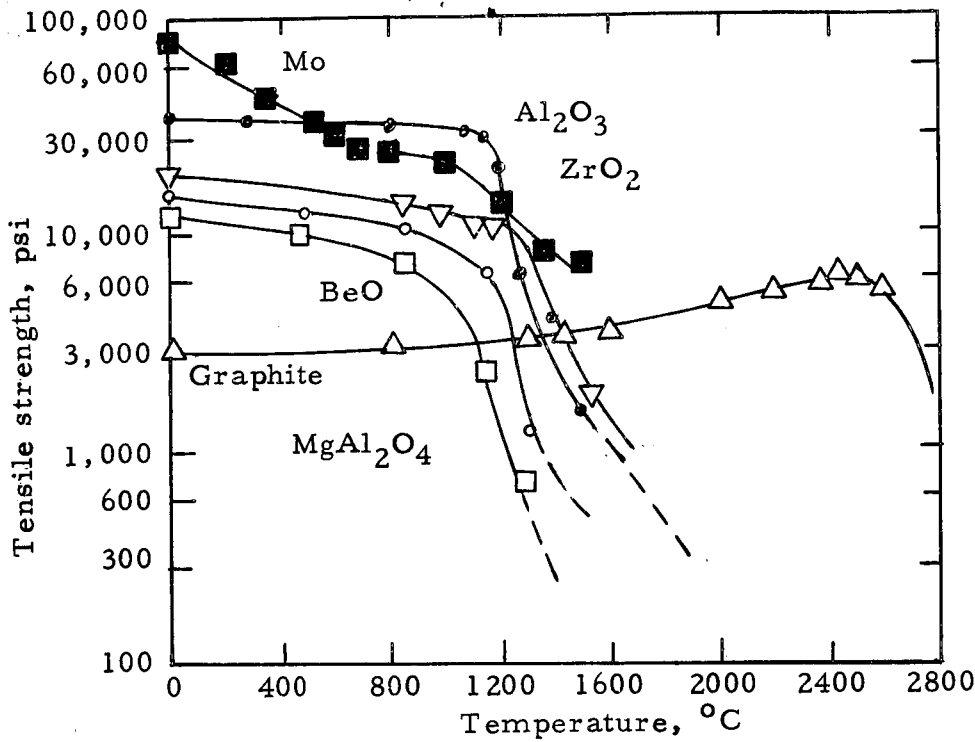Fig. 26

THERMAL EXPANSION AS A FUNCTION OF TEMPERATURE

Fig.18 - High temperature strength of
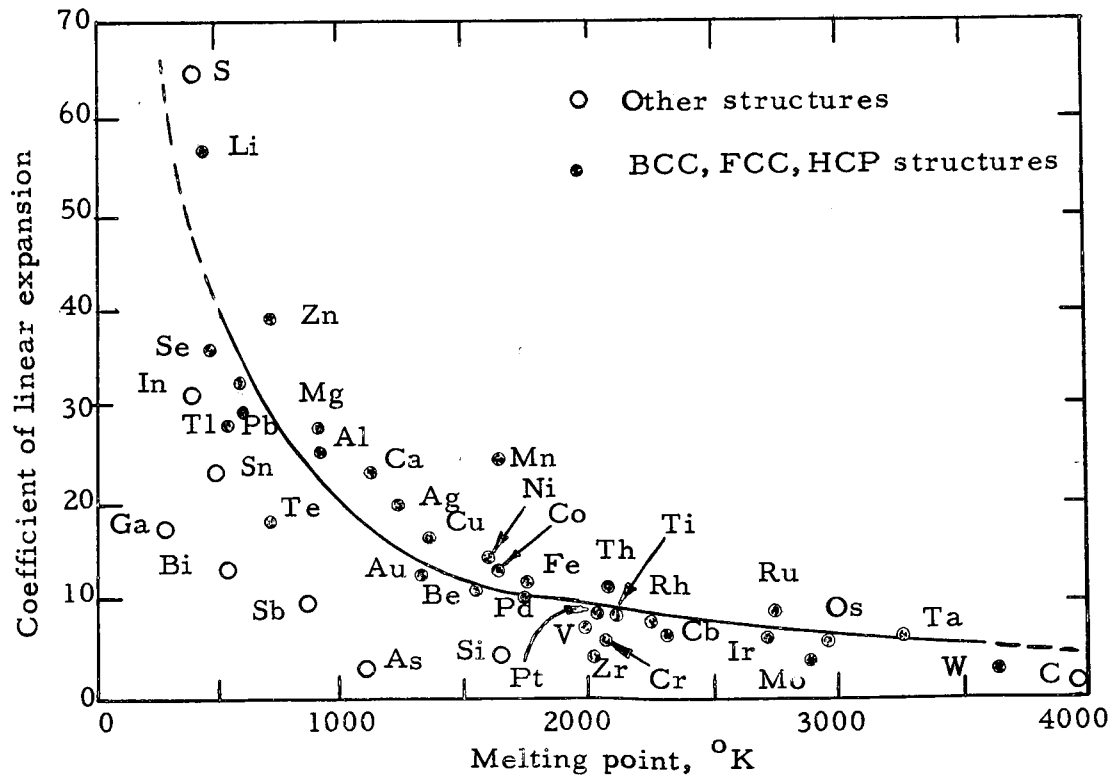several materials.

Fig.27 - Thermal expansion coefficient and melting points
of the elements.